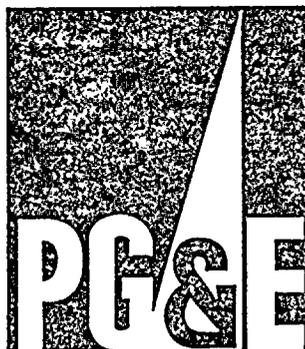


JLS Conversion to Improved Technical Specifications Diablo Canyon Power Plant

Docket # 50-273
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 1.0 - Definitions

ITS 1.0 - Use and Application





IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 1.0

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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(3 Pages)
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METHODOLOGY	(3 Pages)

CROSS-REFERENCE TABLE FOR 1.0
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
1.0			01-01-A	1.1			
1.1			01-01-A	1.1			
1.2			01-01-A	1.1			
1.3			01-30-A 01-32-A	1.1			1.1-1 1.1-9
1.4			01-01-A	1.1			
1.5			01-03-M 01-32-A	1.1			1.1-1
1.6			01-01-A	1.1			
1.7 a. b. c.			01-30-A 01-01-A	1.1			1.1-7
1.8			01-04-A	5.5 3.6	5.5.16 3.6.1		
1.9			01-05-A	3.5.5	3.5.5		
1.10			01-33-A	1.1			
1.11			01-07-A	1.1			
1.12			01-01-A	1.1			
1.13			01-08-A	1.1			1-1-5
1.14			01-13-A	5.5	5.5.1 5.5.4		
1.15			01-09-A	1.4			
1.16			01-31-A			Not Used	
		New	01-10-A	1.1			
1.17			01-11-A	1.1			
1.17 a.1			01-11-A	1.1			
1.17 a.2			01-11-A	1.1			
1.17 a.3			01-11-A	1.1			
1.17 b			01-11-A	1.1			
1.17 c			01-11-A	1.1			
1.18			01-01-A	1.1			
1.19			01-12-A			Not Used	
1.20			01-13-A	5.5	5.5.1		

CROSS-REFERENCE TABLE FOR 1.0
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
1.21			01-01-A	1.1			
1.22			01-14-A 01-25-LS2	1.1			
1.23			01-01-A	1.1			
1.24			01-11-A			Not Used	
1.25			01-16-LG	Moved	FSAR		
1.26			01-15-A			Not Used	
		New	01-17-A	1.1			1.1-6
1.27			01-01-A 01-18-A	1.1			
1.28			01-01-A	1.1			
1.29			01-08-A	1.1			1.1-5
1.30			01-19-A			Not Used	
1.31 a) b)			01-20-M	1.1			
1.32			01-24-A	Moved	FSAR		
1.33			01-01-A	1.1			
1.34		Deleted	N/A			Not Used	
1.35			01-22-A			Not Used	
1.36			01-23-A	1.1			
1.37				1.1			
1.38			01-01-A 01-30-A 01-32-A	1.1			1.1-1
1.39			01-11-A			Not Used	
1.40			01-24-A			Not Used	
1.41			01-15-A			Not Used	
1.42			01-15-A			Not Used	
1.43			01-01-A	1.1			
1.44			01-13-A	5.5	5.5.1 5.5.4		
Table 1.1			01-09-A			Not Used	
Table 1.2			01-25-LS2	Table 1.1-1			1.1-8

CROSS-REFERENCE TABLE FOR 1.0
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
		New	01-26-A	1.2			
		New	01-26-A	1.3			
		New	01-26-A	1.4			1.1-3 1.1-11

CROSS-REFERENCE TABLE FOR 1.0
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
1.0			01-01-A	1.1			
1.1			01-01-A	1.1			
1.2			01-01-A	1.1			
1.4			01-01-A	1.1			
1.5			01-03-M 01-32-A	1.1			1.1-1
1.6			01-01-A	1.1			
1.7			01-01-A 01-30-A	1.1			1.1-7
1.3			01-30-A 01-32-A	1.1			1.1-1 1.1-9
1.10			01-33-A	1.1			
1.43			01-01-A	1.1			
1.11			01-07-A	1.1			
1.12			01-01-A				
1.13			01-08-A				1.1-5
		Not Used					1.1-2
New			01-10-A				
1.17			01-11-A				
3.6.1.2	LCO						
1.18			01-01-A				
1.21			01-01-A				
1.22			01-14-A 01-25-LS2				
1.23			01-01-A				
		New	01-17-A				1.1-6
1.27			01-01-A 01-18-A				
1.28			01-01-A	1.1			
1.29			01-08-A	1.1			1.1-5
1.31			01-20-M	1.1			
1.33			01-01-A	1.1			

CROSS-REFERENCE TABLE FOR 1.0
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
1.36			01-23-A	1.1			
1.37				1.1			
1.38			01-01-A 01-30-A 01-32-A	1.1			1.1-1 1.1-9
Table 1.2			01-25-LS2	Table 1.1-1			1.1-8
		New	01-26-A	1.2			
		New	01-26-A	1.3			
		New	01-26-A	1.4			1-1-3 1.1-11

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

Methodology for Cross-Reference Tables
(Continued)

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).
Note:	The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

**Methodology for Cross-Reference Tables
(Continued)**

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is used to cover all sub-paragraphs (e.g., "A.1" is used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
1.1 - Definitions	1-1
1.2 - Logical Connectors.	1-9a
1.3 - Completion Times	1-9a
1.4 - Frequency	1-9a
Methodology	(2 Pages)

1.0 USE AND APPLICATION

1.01 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

01-01-A

ACTIONS

- 1.1 ACTIONS shall be that part of a Specification which ~~that~~ prescribes remedial ~~measures~~ required actions to be taken under designated conditions within ~~specified completion times~~.

01-01-A

ACTUATION LOGIC TEST

- 1.2 An ACTUATION LOGIC TEST shall be the application of various simulated or ~~actual~~ input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST as a ~~minimum~~ shall include a continuity check as a ~~minimum~~ of output devices.

01-01-A

CHANNEL OPERATIONAL TEST (COT)

- 1.3 A CHANNEL OPERATIONAL TEST (COT) shall be the injection of a simulated or ~~actual~~ signal into the channel as close to the sensor as practicable to verify the OPERABILITY of the required alarm, interlock, including all components in the channel such as alarms, interlocks, displays, and/or trip functions required to perform the specified safety function(s). The COT may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested. The CHANNEL OPERATIONAL TEST (COT) shall include adjustments, as necessary, of the required alarm, interlock and/or trip setpoints such so that the setpoints are within the required range and accuracy.

01-01-A

01-32-A

01-30-A

AXIAL FLUX DIFFERENCE (AFD)

- 1.4 AXIAL FLUX DIFFERENCE (AFD) shall be the difference in normalized flux signals between the top and bottom halves of an ~~two-section~~ excore neutron detector.

01-01-A

CHANNEL CALIBRATION

- 1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such so that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the required sensor, s and alarm, interlock and/or trip functions, those components such as sensors, alarms, displays, and trip functions required to perform the specified safety function(s), and may be performed by any series of sequential, overlapping calibrations or total channel steps such that the entire channel is

01-32-A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

calibrated. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an in-place cross calibration that compares the other sensing elements with the recently installed sensing element.

01-03-M

CHANNEL CHECK

- 1.6 A CHANNEL CHECK shall be the qualitative assessment ~~by observation~~ of channel behavior during operation ~~by observation~~. This determination shall include, where possible, comparison of the channel indication and/or status with ~~to~~ other indications and/or status derived from independent instrument channels measuring the same parameter.

01-01-A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

CHANNEL FUNCTIONAL TEST

1.7 A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels - the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY including required alarm and/or trip functions, or 01:01-A
- b. Bistable channels - the injection of a simulated or actual signal into the sensor to verify OPERABILITY including required alarm and/or trip functions, or 01:01-A
- c. Digital channels - the injection of a simulated or actual signal into the channel as close to the sensor input to the process racks as practical to verify OPERABILITY including required alarm and/or trip functions. 01:01-A

The Channel Functional Test may be performed by means of any series of sequential overlapping, or total channel steps so that the entire channel is tested. 01:30-A

CONTAINMENT INTEGRITY

01:04-A

~~1.8 CONTAINMENT INTEGRITY shall exist when:~~

- ~~a. All penetrations required to be closed during accident conditions are either:
 - ~~1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or~~
 - ~~2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.~~~~
- ~~b. All equipment hatches are closed and sealed,~~
- ~~c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,~~
- ~~d. The containment leakage rates are within the limits of specification 3.6.1.2, and~~
- ~~e. The sealing mechanism associated with each penetration (e.g., welds, bellows or O rings) is OPERABLE.~~

CONTROLLED LEAKAGE

01:05-A

~~1.9 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.~~

1.0 USE AND APPLICATION

1.01 DEFINITIONS

CORE ALTERATIONS

- 1.10 CORE ALTERATIONS shall be the movement or manipulation of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

01-33A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

DOSE EQUIVALENT I-131

- 1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present.

The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Dose Factors for Power and Test Reactor Sites" or those listed in Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

01-07-A

E - AVERAGE DISINTEGRATION ENERGY

- 1.12 \bar{E} shall be the average (weighted in proportion to the concentration of each radionuclide in the sample reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half-lives greater than 10 minutes making up at least 95% of the total non-iodine activity in the coolant.

01-01-A

ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME

- 1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be verified by means of any series of sequential overlapping or total steps so that the entire response time is verified.

01-08-A

ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURE

- 1.14 The ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURE (ERMP) shall contain a description of sample locations, types of sample locations, methods and frequency of analysis, and reporting requirements.

01-13-A

FREQUENCY NOTATION

- 1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

01-09-A

GASEOUS RADWASTE SYSTEM

- 1.16 A GASEOUS RADWASTE SYSTEM shall be any system designed and installed to

01-31-A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

~~reduce radioactive gaseous effluents by collecting Reactor Coolant System off-gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~

(new) L_1 The maximum allowable primary containment leakage rate, L_1 , shall be 0.10 % of primary containment air weight per day at the calculated peak containment pressure (P_1)

1-10-A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

IDENTIFIED LEAKAGE shall be:

1.17 a. ~~IDENTIFIED~~ ~~Identified~~ LEAKAGE shall be:

- 1 a. ~~Leakage, except CONTROLLED LEAKAGE, into closed systems, LEAKAGE such as that from pump seals or valve packing leaks (except reactor coolant pump (RCP) seal water injection or leakoff) that are captured and conducted to a collection systems or a sump or collecting tank; or~~ 01-11-A
- 2 b. ~~Leakage~~ LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be ~~PRESSURE BOUNDARY~~ ~~pressure boundary~~ LEAKAGE; or
- 3 e. ~~Reactor Coolant System leakage (RCS) LEAKAGE through a steam generator (SG) to the Secondary Coolant System.~~

b. Unidentified LEAKAGE

~~All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE.~~ 01-11-A

c. Pressure Boundary LEAKAGE

~~LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.~~

MASTER RELAY TEST

1.18 A MASTER RELAY TEST shall be ~~the energization of~~ ~~consist of energizing~~ each master relay and ~~verification of~~ ~~verifying the~~ OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay. 01-01-A

MEMBER(S) OF THE PUBLIC

1.19 ~~MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.~~ 01-12-A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

OFFSITE DOSE CALCULATION PROCEDURE

01-13-A

~~1.20 The OFFSITE DOSE CALCULATION PROCEDURE (ODCP) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints.~~

OPERABLE - OPERABILITY

1.21 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electric power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified safety function(s) are also capable of performing their related support function(s).

01-01-A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

OPERATIONAL MODE — MODE

1.22 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature and reactor vessel head closure bolt tensioning specified in Table 1-2 1 1-1 with fuel in the reactor vessel

01-14-A

PHYSICS TESTS

1.23 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Nuclear Regulatory Commission.

01-25-LS2

01-01-A

PRESSURE BOUNDARY LEAKAGE

~~1.24 PRESSURE BOUNDARY LEAKAGE shall be leakage, except steam generator tube leakage, through a non isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.~~

01-11-A

PROCESS CONTROL PROGRAM

~~1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.~~

01-16-LG

PURGE — PURGING

~~1.26 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.~~

01-15-A

(new)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and enable temperature associated with the Low Temperature Over pressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with ITS Specification 5.6.6. Plant operation within these operating limits is addressed in individual specifications.

01-17-A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

QUADRANT POWER TILT RATIO (QPTR)

- 1.27 ~~QUADRANT POWER TILT RATIO QPTR~~ shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. ~~With one excore detector inoperable, the remaining three detectors shall be used for computing the average.~~
- 01-01-A
- 01-18-A

RATED THERMAL POWER (RTP)

- 1.28 ~~RATED THERMAL POWER RTP~~ shall be a total reactor core heat transfer rate to the reactor coolant of 3338 Mwt for Unit 1 and 3411 Mwt for Unit 2.
- 01-01-A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.29 The REACTOR TRIP SYSTEM RTS RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be verified by means of any series of sequential, overlapping, or total steps so that the entire response time is verified.

01-08-A

REPORTABLE EVENT

~~1.30 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.~~

01-19-A

SHUTDOWN MARGIN (SDM)

1.31 SHUTDOWN MARGIN SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

- a) All full-length rod cluster control assemblies (RCCAs) (shutdown and control) are fully inserted except for the single rod cluster assembly (RCCA) of highest reactivity worth which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and
- b) In MODES 1 and 2, the fuel and moderator temperatures are changed to the hotzero power temperatures.

01-20-M

01-20-M

SITE BOUNDARY

~~1.32 SITE BOUNDARY shall be that line as shown in Figure 5.1-3.~~

01-24-A

SLAVE RELAY TEST

1.33 A SLAVE RELAY TEST shall be the energization of consist of energizing each slave relay and verification of verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check, as a minimum, of associated testable actuation devices.

01-01-A

1.34 Deleted

1.0 USE AND APPLICATION

1.01 DEFINITIONS

SOURCE CHECK

~~1.35 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.~~

01-22-A

STAGGERED TEST BASIS

1.36 A STAGGERED TEST BASIS shall consist of ~~the testing of one of the~~

- a. ~~A test schedule for n systems, subsystems, trains channels or other designated components obtained by dividing the specified test interval into n equal sub-intervals, and during the interval specified by the surveillance frequency, so that all systems, subsystems, channels, or other designated components are tested during n surveillance frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.~~
- b. ~~The testing of one system, subsystem, train or other designated component at the beginning of each sub-interval.~~

01-23-A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

THERMAL POWER

1.37 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)

1.38 A ~~TRIP ACTUATING DEVICE OPERATIONAL TEST TADOT~~ shall consist of operating the ~~7~~ trip A actuating ~~0~~ device and verifying OPERABILITY, including all components in the channel, such as alarms, interlocks, displays and/or trip functions required to perform the specified safety function(s). ~~The TADOT may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested. The TRIP ACTUATING DEVICE OPERATIONAL TEST TADOT shall include adjustment, as necessary, of the 7 trip A actuating 0 device such so that it actuates at the required setpoint within the required accuracy.~~

01-01-A

01-32-A

01-30-A

UNIDENTIFIED LEAKAGE

~~1.39 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.~~

01-11-A

UNRESTRICTED AREA

~~1.40 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.~~

01-24-A

VENTILATION EXHAUST TREATMENT SYSTEM

~~1.41 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not normally considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.~~

01-15-A

1.0 USE AND APPLICATION

1.01 DEFINITIONS

VENTING

~~1.42 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.~~

01-15-A

CORE OPERATING LIMITS REPORT (COLR)

~~1.43 The CORE OPERATING LIMITS REPORT COLR is the unit specific document that provides core operating cycle specific parameter limits for the current operating reload cycle. These cycle specific core operating parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.1.8 5.6.5. Unit Plant operation within these operating limits is addressed in individual specifications.~~

01-01-A

RADIOLOGICAL MONITORING AND CONTROLS PROGRAM

~~1.44 The RADIOLOGICAL MONITORING AND CONTROLS PROGRAM (RMCP) shall contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.5 and 6.9.1.6.~~

01-13-A

TABLE 1.1
FREQUENCY NOTATION

01-09-A

NOTATION

FREQUENCY

S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Completed prior to each release.
N.A.	Not applicable.

TABLE 1-2
OPERATIONAL MODES

01-25-LS2

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*(a)	AVERAGE REACTOR COOLANT TEMP. (°F)
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 350^{\circ}\text{F}$ NA
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 350^{\circ}\text{F}$ NA
3. HOT STANDBY	< 0.99	\emptyset NA	$\geq 350^{\circ}\text{F}$
4. HOT SHUTDOWN (b)	< 0.99	\emptyset NA	$350^{\circ}\text{F} > T_{avg}$ $> 200^{\circ}\text{F}$
5. COLD SHUTDOWN (b)	< 0.99	\emptyset NA	$\leq 200^{\circ}\text{F}$
6. REFUELING** (c)	≤ 0.95 NA	\emptyset NA	$\leq 140^{\circ}\text{F}$ NA

*(a) Excluding decay heat.

** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

(b) the required reactor vessel head closure bolts fully tensioned

(c) the required reactor vessel head closure bolts less than fully tensioned

(new) NUREG-1431 Sections

01-26-A

1.2. Logical Connectors

1.3. Completion Times and

1.4. Frequency

are new sections which will be incorporated in the IIS in their entirety. See the corresponding sections in Enclosure 5A.

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. Deletions - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. Additions - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. Modifications - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. Administrative - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

**Methodology For Mark-Up of Current TS
(Continued)**

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers

None

Description of Changes

(5 Pages)

DESCRIPTION OF CHANGES TO TS SECTION 1.0

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	A	These definitions would be reworded to be consistent with NUREG-1431. The proposed rewording included in this category does not involve any changes of a technical nature.
01-02	A	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-03	M	The definition of CHANNEL CALIBRATION is reworded to be consistent with NUREG-1431. The revised wording provides additional detail concerning calibration of instrument channels with resistance temperature detector (RTDs) or thermocouples.
01-04	A	The definition of CONTAINMENT INTEGRITY would no longer be used and the specifications in ITS Section 3.6 and the Administrative Controls Section would be revised accordingly. The CTS definition for CONTAINMENT INTEGRITY would be deleted to be consistent with NUREG-1431. This definition is effectively incorporated into the NUREG-1431 Bases for the new Containment Limiting Condition for Operation (LCO) (ITS 3.6.1) and the Administrative Controls Section for the Containment Leakage Testing Program [].
01-05	A	The current definition for CONTROLLED LEAKAGE would be in accordance with NUREG-1431. This definition will no longer be required for the ITS because LCO 3.5.5 ensures that reactor coolant pump (RCP) seal injection flow remains within limits. Therefore, this change is not technical and has been categorized as administrative.
01-06	LS1	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B)
01-07	A	The location of the thyroid dose conversion factors used for the calculation of DOSE EQUIVALENT I-131 have been added in accordance with NUREG-1431.

DESCRIPTION OF CHANGES TO TS SECTION 1.0

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-08	A	The CTS definitions for ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME and REACTOR TRIP SYSTEM (RTS) RESPONSE TIME would be modified to be consistent with NUREG-1431. In addition, the term "measured" would be replaced by "verified" to be consistent with the requirements of ITS Surveillance Requirement (SR) 3.3.1.16 and SR 3.3.2.10 to verify response time is within limits. The addition of the statement that response time may be verified by means of any series of sequential, overlapping, or total steps so that the entire response time is verified, is administrative in nature. This is consistent with the methodology presently described in the CTS Bases for demonstrating total channel response time.
01-09	A	The CTS definition for FREQUENCY NOTATION (and Table 1.1, FREQUENCY NOTATION) would be deleted to be consistent with NUREG-1431. The acronyms defined in Table 1.1, FREQUENCY NOTATION, are no longer used in NUREG-1431. Surveillance frequencies are specified in NUREG-1431; thereby, obviating the definition. This is a nontechnical change made to conform to NUREG-1431.
01-10	A	The definition for maximum allowable primary containment leakage rate (L_p) would be added in the ITS to be consistent with NUREG-1431. This addition has been determined to be an administrative change on the basis that this definition has simply been [copied] from the [CTS Administrative Controls 6.8.4.j] to the definitions.
01-11	A	The CTS definitions for IDENTIFIED LEAKAGE, UNIDENTIFIED LEAKAGE, and PRESSURE BOUNDARY LEAKAGE have been merged into one definition for LEAKAGE and reworded to be consistent with NUREG-1431. This is a nontechnical change since it will not alter the manner in which LEAKAGE is accounted for and treated from present practice. The definition of UNIDENTIFIED LEAKAGE has been expanded to include "except RCP seal water [injection or leakoff,]" to be consistent with NUREG-1431.
01-12	A	The CTS definition for MEMBER OF THE PUBLIC, would be deleted to be consistent with NUREG-1431. This definition would be deleted on the basis that it is defined in 10 CFR 20.1003 and 40 CFR 190.
01-13	A	The CTS definition of the Offsite Dose Calculation Manual (ODCM) [OFFSITE DOSE CALCULATION PROCEDURE (ODCP), ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURE (ERMP), and RADIOLOGICAL MONITORING AND CONTROLS PROGRAM (RMCP)] would be [merged where duplications occur and would be] incorporated into the Administrative Controls Section 5.5.1 of the ITS. [These] changes [are] nontechnical because the definitions of the ODCM [ODCP, ERMP, AND RMCP] will be [combined and] moved to another section of the ITS.

DESCRIPTION OF CHANGES TO TS SECTION 1.0

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-14	A	The CTS definition for OPERATIONAL MODE would be revised to "MODE" and the wording would be revised to be consistent with NUREG-1431. The changes are nontechnical since they will not affect current practice.
01-15	A	The CTS definitions of heating, ventilation, and air conditioning (HVAC) systems and functions would be deleted to be consistent with NUREG-1431. "[Ventilation Exhaust Treatment System,]" "PURGE -SPARE PURGING," and "VENTING," where used, do not require special definitions.
01-16	LG	The CTS definition of the PROCESS CONTROLS PROGRAM (PCP) would be moved outside of the TS along with the Administrative Controls description (CTS 6.8.1.e) of this program to be consistent with NUREG-1431. The PCP definition and program description from Administrative Controls are moved into the Final Safety Analysis Report (FSAR). The PCP implements regulatory requirements and need not be restated in the TS. The requirement to comply with applicable Federal and State regulations for the processing of radioactive waste provides sufficient control of future changes to the PCP.
01-17	A	The definition of a PRESSURE TEMPERATURE LIMITS REPORT (PTLR) would be added to be consistent with NUREG-1431 and Westinghouse Owners Group (WOG) 67, Rev. 1. The definition will support the use of a PTLR. Adding the definition is administrative in nature.
01-18	A	The portion of the QUADRANT POWER TILT RATIO (QPTR) definition dealing with an inoperable excore detector is addressed in the Conditions and SRs of ITS 3.2.4.
01-19	A	The CTS definition of REPORTABLE EVENT is not used in the ITS and would be deleted to be consistent with NUREG-1431. This definition would be deleted on the basis that a REPORTABLE EVENT is defined by 10 CFR 50.72 and 50.73. This change is administrative in nature because it will have no effect on current reporting practices.
01-20	M	The CTS definition of SHUTDOWN MARGIN (SDM) would be revised to be consistent with NUREG-1431. The requirement to account for any rod control cluster assemblies (RCCAs) not capable of being fully inserted was moved from CTS ACTION and SRs. The only substantive technical change to this definition is the addition of the requirement that in MODES 1 and 2, the fuel and moderator temperatures be changed to the hot zero power temperatures. This ensures that the power defect due to shutting the reactor down from MODES 1 or 2 is accounted for in the SDM. While this requirement is consistent with current practice, it has not been specified in the existing definition. Consequently, it has been categorized as a more restrictive change.

DESCRIPTION OF CHANGES TO TS SECTION 1.0

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-21		Not used.
01-22	A	The definition of SOURCE CHECK is deleted from the CTS in accordance with NUREG-1431. No surveillances in the ITS require SOURCE CHECKS; therefore, this is an administrative change. It will be defined where used in licensee controlled documents; however, it has not been used in the CTS since the implementation of NRC Generic Letter 89-01.
01-23	A	The CTS definition for STAGGERED TEST BASIS (STB) would be revised to be consistent with NUREG-1431. The test intervals for throughout the ITS that are to be performed on a STB will be revised to be consistent with the new definition so that there will be no net change in the CTS implementation of staggered test intervals. For example, under the CTS, if a parameter is monitored by 3 channels of instrumentation, and the test interval is quarterly, 1 channel would be tested each month during any given quarter by dividing the test interval into 3 equal sub-intervals. Under the new definition, the test interval for that same instrumentation in the ITS would be specified as monthly so that the net effect is the same. One channel would be tested each month during any given quarter.
01-24	A	The CTS definitions of SITE BOUNDARY and UNRESTRICTED AREA ARE deleted to be consistent with NUREG-1431. These definitions are deleted on the basis that they are defined in 10 CFR 20.1003.
01-25	LS2	Table 1.2 of the CTS would become Table 1.1-1 in the ITS. The following changes would be made to conform to NUREG-1431. In ITS Table 1.1-1, the notation "NA" would replace "0" under % RATED THERMAL POWER for MODES 3, 4, 5, and 6. This is a nontechnical change since with K_{eff} less than 0.99, THERMAL POWER would be zero anyway. For MODE 6, the temperature has been replaced with NA since there is no safety analysis basis for the value of 140°F specified in the CTS. Also for MODE 6, the reactivity Condition has been designated NA since the value of 0.95 is specified in the Bases of ITS 3.9.1. The temperatures for MODES 1 and 2 are designated as NA on the basis that temperature for these MODES is less restrictive than the minimum temperature for criticality and the operating program for reactor coolant system Tavg. A new Note b has been added to MODES 4 and 5 stating that the required number of reactor vessel head closure bolts are fully tensioned, and a new Note c replaces the Note applied to MODE 6. The new Note c states that the required reactor vessel head closure bolts are less than fully tensioned. The new Note c no longer specifies that fuel is in the vessel because the condition of fuel in the vessel is addressed by the definition of the term MODE. This definition stipulates that fuel be in the vessel in order to be in a "MODE." These changes are administrative, except for the new Notes b and c per Traveler TSTF-88 and addressed in NSHC LS2.

DESCRIPTION OF CHANGES TO TS SECTION 1.0

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-26	A	New Sections 1.2, 1.3, and 1.4 would be incorporated into the ITS to be consistent with NUREG-1431. Section 1.2 provides specific examples of the use of the logical connectors <u>AND</u> and <u>OR</u> and the numbering sequence associated with their use in the ITS. Section 1.3 deals with the proper use and interpretation of Completion Times, and specific examples are given that will aid the user in understanding Completion Times. Section 1.4 deals with the proper use and interpretation of surveillance Frequencies. Specific examples are given that will aid the user in understanding surveillance Frequencies as they will appear in the ITS. The proposed changes are administrative in nature and by themselves are not technical changes, incorporating Travelers WOG-74, Rev. 1, and WOG-90, Rev. 1.
01-27	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-28	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-29	LS3	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-30	A	Consistent with TSTF-39, Rev. 1, the definitions of COT, [CHANNEL FUNCTIONAL TEST (CFT)], and TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) are expanded to include the details of acceptable performance methodology. Performance of these tests in a series of sequential, overlapping, or total channel steps provides the necessary assurance of appropriate operation of the entire channel. This change also makes the COT, [CFT], and TADOT definitions consistent with the CTS and the NUREG-1431 definition of CHANNEL CALIBRATION which already contains similar wording.
01-31	A	Definitions of specific plant systems which are defined by the plant design are deleted consistent with NUREG-1431. The definitions contained in ITS 1.0 are intended for definitions that are necessary for the understanding of the specifications and can be generically defined for most plants. Definitions of systems that are not used in the specifications, or are specific to a particular plant (or only a few plants) are no longer defined in this section. Where necessary, such items are defined in the Bases for the applicable specifications.
1-32	A	The definitions of CHANNEL CALIBRATION, COT, [CFT], and TADOT are reworded to be consistent with Industry Traveler TSTF-64 to clarify the phrase "entire channel," thus reducing the potential for inconsistent interpretation of the phrase as experienced by a number of plants.
1-33	A	This change revises the CTS definition of CORE ALTERATIONS to delete "or manipulation" and "conservative" in accordance with NUREG-1431. The words as used in the definition were redundant and deleting the words does not alter the meaning of the definition.

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(4 pages)

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	These definitions would be reworded to be consistent with NUREG-1431. The proposed rewording included in this category does not involve any changes of a technical nature.	Yes	Yes	Yes	Yes
01-02 A	The definitions for analog COT and digital COT would be combined into a single definition of COT in accordance with NUREG-1431.	No, DCPD CTS does not include the digital COT definition.	Yes	No, do not have the digital COT definition.	No, "Digital" is not included in CTS.
01-03 M	The definition of CHANNEL CALIBRATION is reworded. The revised wording provides additional detail concerning calibration of instrument channels with RTDs or thermocouples.	Yes	Yes	Yes	Yes
01-04 A	This definition would no longer be used and the specifications in Section 3.6 and Administrative Controls would be revised accordingly. The CTS definition for CONTAINMENT INTEGRITY would be relocated to the Administrative Controls Section.	Yes	Yes	Yes	Yes, see also ITS 5.5.6 and 5.5.16.
01-05 A	The CTS definition for CONTROLLED LEAKAGE would be deleted. The definition is not required because ITS LCO 3.5.5 ensures that RCP seal injection flow remains within limits.	Yes	Yes	No, see Change Number 01-28-LG.	No, see Change Number 01-28-LG.
01-06 LS1	The CTS definition for CORE ALTERATIONS would be modified to qualify a CORE ALTERATION as movement of fuel, sources, or other reactivity control components.	No, this definition is included in the DCPD CTS.	Yes	Yes	Yes
01-07 A	The location of the thyroid dose conversion factors used for the calculation of DOSE EQUIVALENT I-131 have been added.	Yes	No, already in CTS.	No, already in CTS.	No, already in CTS.

CONVERSION COMPARISON TABLE - CURRENT TS 1.0

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-10 A	The CTS Administrative Controls Section definition for maximum allowable primary containment leakage rate (L_p) would be added to the ITS.	Yes	Yes	Yes	Yes
01-11 A	The CTS definitions for IDENTIFIED LEAKAGE, UNIDENTIFIED LEAKAGE, and PRESSURE BOUNDARY LEAKAGE have been merged into one definition for LEAKAGE and reworded.	Yes	Yes	Yes	Yes
01-12 A	The CTS definition for MEMBER OF THE PUBLIC, which is defined in 10 CFR 20.1003, would be deleted.	Yes	Yes	Yes	Yes
01-13 A	The CTS definitions of the (ODCM), [RMCP, and ERMP] would be moved to the Administrative Controls Section of the ITS.	Yes	Yes	Yes	Yes
01-14 A	The CTS definition of 'OPERATIONAL MODE' would be revised to 'MODE' and reworded.	Yes	Yes	Yes	Yes
01-15 A	The CTS definitions of HVAC systems and functions would be deleted. ["Ventilation and Exhaust Treatment System,]" "PURGE - PURGING," and "VENTING," where used, do not require special definitions.	Yes	Yes	Yes	Yes
01-16 LG	The CTS definition of the PCP would be moved outside of the TS along with the Administrative Controls description of this program.	Yes, moved to the FSAR.	Yes, moved to the FSAR	Yes, moved to USAR.	Yes, moved to FSAR Section 16.25.
01-17 A	The definition of a PTLR would be added to support the use of a PTLR.	Yes	Yes	Yes	Yes
01-18 A	The portion of the QPTR definition dealing with an inoperable excore detector is addressed in the CONDITIONS and SRs of ITS 3.2.4.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 1.0

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-19 A	The CTS definition of REPORTABLE EVENT is not used in the ITS and is deleted.	Yes	Yes	Yes	Yes
01-20 M	The CTS definition of SDM is revised. The requirement to account for any RCCAs not capable of being fully inserted was simply moved from CTS ACTION and SRs. The only substantive technical change to this definition is the addition of the requirement that in MODES 1 and 2, the fuel and moderator temperatures be changed to the hot zero power temperatures.	Yes	Yes	Yes	Yes
01-21	Not used.	NA	NA	NA	NA
01-22 A	The definition of SOURCE CHECK is deleted from the CTS since it is not used in NUREG-1431.	Yes	Yes	Yes	Yes
01-23 A	The CTS definition for STB would be revised. The test intervals for SRs throughout the ITS that are to be performed on a STB will be revised to be consistent with the new definition.	Yes	Yes	Yes	Yes
01-24 A	The CTS definitions of SITE BOUNDARY and UNRESTRICTED AREA which are defined in 10 CFR 20.1003 would be deleted.	Yes	Yes	Yes	Yes
01-25 LS2	Table 1.2 of the CTS would become Table 1.1-1 in the ITS. Several changes would be made to conform to NUREG-1431. (e.g., ITS Table 1.1-1, the notation "NA" would replace "O" under % RTP for MODES 3, 4, 5, and 6.) Reactor vessel head closure bolt tensioning is revised per Traveler TSTF-88 and is discussed in NSHC LS2.	Yes	Yes	Yes	Yes
01-26 A	New Sections 1.2, 1.3, and 1.4 would be incorporated into the ITS	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 1.0

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-27 M	The definition of RAFDO is deleted.	No	No	No	Yes, definition only in Callaway CTS.
01-28 LG	The definition of CONTROLLED LEAKAGE is deleted. The RCP seal water return flow limit is moved to a licensee controlled document.	No, see change number 01-05-A.	No, see Change Number 01-05-A.	Yes, moved to USAR Section 16.	Yes, moved to FSAR Section 16.4.
01-29 LS3	Allows measuring QPTR when one or more excore detector channels are inoperable with moveable in-core detectors.	No	Yes, portion of the definition being changed is only in the CPSES CTS.	No	No
01-30 A	The definitions of COT, [CFT], and TADOT are expanded to include the details of acceptable performance methodology. Performance of these tests in a series of sequential, overlapping, or total channel steps provides the necessary assurance of appropriate operation of the entire channel.	Yes	Yes	Yes	Yes
01-31 A	Definitions of specific plant systems which are defined by the plant design are deleted.	Yes	Yes	No, not in CTS	No, not in CTS
01-32 A	The definition of CHANNEL CALIBRATION, COT, [CFT] and TADOT are reworded to be consistent with Industry Traveler TSTF-64. The revised wording clarifies what is meant by "entire channel."	Yes	Yes	Yes	Yes
01-33 A	This change revises the CTS definition of CORE ALTERATIONS to delete "or manipulation" and "conservative."	Yes	Yes	Yes	Yes

ENCLOSURE 4
NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

PAGE

I. Organization 2

II. Description of NSHC Evaluations. 3

III. Generic NSHCs

 "A" - Administrative Changes 5

 "R" - Relocated Technical Specifications 7

 "LG" - Less Restrictive (moving information out of the TS) 10

 "M" - More Restrictive 12

IV. Specific NSHCs - "LS"

 LS1 (not applicable to DCP)

 LS2 13 14

 LS3 (not applicable to DCP)

I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3..." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2

10CFR50.92 EVALUATION
FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

CTS Table 1.2 (ITS Table 1.1-1) is revised such that the required vessel head closure bolt requirements for MODES 4, 5, and 6 are clarified. Currently a footnote applicable only to MODE 6 defines that MODE, in part, by reference to "vessel head closure bolts less than fully tensioned." That footnote does not specify the transition point between MODES 5 and 6 with regard to the number of vessel head closure bolts that must be fully tensioned, leaving the issue open to interpretation. The proposed change provides the necessary clarification by adding a footnote to MODES 4 and 5, consistent with the approach used in NUREG-1431 to define those MODES as having the required number of reactor head closure bolts fully tensioned. The transition point between MODES 5 and 6 would also be clarified as occurring when the reactor vessel head closure bolts are less than fully tensioned. The required number of closure bolts, which may be less than the total number, is established by analysis that demonstrates adequate O-ring compression to prevent leakage and ensures that ASME Section III stress limits for affected components are not exceeded. This revision is consistent with TSTF-88.

The proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change clarifies the requirement for one or more required reactor vessel head closure bolts not fully tensioned as a condition to define MODE 6. The proposed change would not result in any hardware changes, would not affect the initiators of any analyzed events, and would not alter assumptions relative to mitigation of accident or transient events. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2 (continued)

The proposed change does not involve any changes in the method by which any safety-related system performs its function. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. There is no effect on systems necessary to assure the accomplishment of accident mitigation. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS2" resulting from the conversion to the ITS format are seen to satisfy the NSHC of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers	(1 Page)
NUREG-1431 Specifications that are not applicable	(1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
1.1	1.1-1
1.2	1.2-1
1.3	1.3-1
1.4	1.4-1
Methodology	(2 Pages)

Industry Travelers Applicable to Section 1.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-19, Rev. 1	Not Incorporated	NA	Not NRC approved as of traveler cut-off date.
TSTF-39, Rev. 1	Incorporated	1.1-9	
TSTF-64	Incorporated	1.1-1	
TSTF-88	Incorporated	1.1-8	
TSTF-111, Rev. 1	Incorporated	1.1-5	
WOG-67, Rev. 1	Incorporated	1.1-6	
WOG-74, Rev. 1	Incorporated	1.1-3	
WOG-90, Rev. 1	Incorporated	1.1-11	

NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

Specification #

Specification Title

Comments

NONE

1.0 USE AND APPLICATION

1.1 Definitions

NOTE

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between <u>B-PS</u> the top and bottom halves of a two-section an excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass those components the entire channel, including the required such as sensors, alarms, interlock, displays, and trip functions required to perform the specified safety function(s). Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an in-place cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated. <u>1.1-1</u>

(Continued)

1.1 Definitions (continued)

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST (CFT)

A CFT shall be:

- a. Analog channels - the injection of a simulated or actual signal into the channel as close to the sensor as practical to verify OPERABILITY including required alarm and trip functions. 1.1-7
- b. Bistable channels - the injection of a simulated or actual signal into the sensor to verify OPERABILITY including required alarm and trip functions. 1.1-7
- c. Digital channels - the injection of a simulated or actual signal into the channel as close to the sensor input to the process racks as practical to verify OPERABILITY including required alarm and trip functions. 1.1-7

The Channel Functional Test may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested. 1.1-9

CHANNEL OPERATIONAL

A COT shall be the injection of a simulated or actual signal TEST (COT) into the channel as close to the sensor as practicable to verify the OPERABILITY of required including all components in the channel such as alarms, interlocks, displays, and trip functions required to perform the specified safety function(s). The COT may be performed by means of any series of sequential, overlapping or total channel steps so that the entire channel is tested. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy. 1.1-1
1.1-9

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

(Continued)

1.1 Definitions (continued)

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of NRC Regulatory Guide 1.109, Rev. 1, October, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity" .]~~

B

\bar{E} -AVERAGE DISINTEGRATION ENERGY

\bar{E} shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > ~~[15]~~ minutes, making up at least 95% of the total non-iodine activity in the coolant.

B-PS

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured ~~verified~~ by means of any series of sequential, overlapping, or total steps so that the entire response time is measured ~~verified~~.

1.1-5

L_p

The maximum allowable primary containment leakage rate, L_p , shall be ~~[0.10]~~ % of primary containment air weight per day at the calculated peak containment pressure (P_p).

B-PS

(Continued)

1.1 Definitions (continued)

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or
3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System.

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE-OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required

(Continued)

1.1 Definitions (continued)

for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

- a. Described in Chapter ~~[14 Initial Test Program]~~ of the FSAR;
- b. Authorized under the provisions of 10 CFR 50.59; or
- c. Otherwise approved by the Nuclear Regulatory Commission.

B-PS

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in individual specifications, LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

1.1-6

QUADRANT POWER TILT RATIO (QPTR)

QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~[2893 Mwt]~~ [3338 MWT for Unit 1 and 3411 MWT for Unit 2]

B-PS

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured verified by means of any series of sequential, overlapping, or total steps so that the entire response time is measured verified.

1.1-5

(Continued)

1.1 Definitions (continued)

SHUTDOWN MARGIN (SDM)	<p>SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:</p> <p>a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and</p> <p>b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal hot zero power design level temperatures]</p>	<hr/> <p>B-PS</p>
SLAVE RELAY TEST	<p>A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.</p>	
STAGGERED TEST BASIS	<p>A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.</p>	
THERMAL POWER	<p>THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.</p>	
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	<p>A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required including all components, such as alarms, interlocks, displays, and trip functions required to perform the specified safety function(s). The TADOT may be performed by means of any series of sequential overlapping or total channel steps so that the entire channel is tested. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.</p>	<hr/> <p>1.1-1</p> <hr/> <p>1.1-9</p> <hr/>

(Continued)

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER ^(a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	$\geq [350]$
4	Hot Shutdown ^(b)	< 0.99	NA	$[350] > T_{avg} > [200]$
5	Cold Shutdown ^(b)	< 0.99	NA	$\leq [200]$
6	Refueling ^(c)	NA	NA	NA

B
B
B

(a) Excluding decay heat.

(b) All ~~The required~~ reactor vessel head closure bolts fully tensioned.

1.1-8

(c) ~~One or more The required~~ reactor vessel head closure bolts less than fully tensioned.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES The following examples illustrate the use of logical connectors.

(Continued)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify... <u>AND</u> A.2 Restore...	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(Continued)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip... <u>OR</u> A.2.1 Verify... <u>AND</u> A.2.2.1 Reduce... <u>OR</u> A.2.2.2 Perform... <u>OR</u> A.3 Align...	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

1.3 Completion Times

DESCRIPTION
(continued)

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

- a. Must exist concurrent with the first inoperability; and
- b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

- a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or
- b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery..." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

(Continued)

1.3 Completion Times

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

(Continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

(Continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(Continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. <u>AND</u> One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. <u>OR</u> C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

(Continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock". In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

(Continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

(Continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-5

ACTIONS

NOTE

Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(Continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

(Continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x.	Once per 8 hours
	<u>OR</u> A.2 Reduce THERMAL POWER to \leq 50% RTP.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of

Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(Continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time

(Continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE
COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

EXAMPLES The following examples illustrate the various ways that Frequencies are specified. In these examples 1 through 4, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3. In example 5, the Applicability of the LCO is MODES 1, 2, 3, and 4.

(Continued)

1.4 . Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

(Continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP <u>AND</u> 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(Continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>NOTE</p> <p>Not required to be performed until 12 hours after \geq 25% RTP.</p> <hr style="border-top: 1px dashed black;"/> <p>Perform channel adjustment.</p>	<p>7 days</p>

The interval continues, whether or not the unit operation is $<$ 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is $<$ 25% RTP, this Note allows 12 hours after power reaches \geq 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was $<$ 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power \geq 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(Continued)

1.4 Frequency

EXAMPLES (continued) EXAMPLE 1.4-4

1.1-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
NOTE	
Only required to be performed in MODE 1	
Perform complete cycle of the valve	7 days

The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance have been performed. If the Surveillance were not performed upon prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply (as well as having had a violation of SR 3.0.4).

(Continued)

1.4 Frequency

EXAMPLES (continued) EXAMPLE 1.4-5

1.1-11

SURVEILLANCE REQUIREMENTS

1.1-3

SURVEILLANCE	FREQUENCY
Verify each containment isolation manual valve is closed	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days

In Example 1.4-5, the "specified Frequency" begins when the Surveillance is performed, but when the interval expires the Surveillance is not required to be performed until certain conditions are met. The Surveillance must be performed prior to entering MODE 4 from MODE 5, but only if the 92 day "specified Frequency" has passed. Although the period prior to the specified conditions is given as 92 days, the time interval may be extended to 1.25 times the stated period as allowed by SR 3.0.2 for operational flexibility.

The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the conditions in the Frequency are met and the interval specified by SR 3.0.2 is exceeded without the Surveillance having been performed and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes- Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

**Methodology For Mark-up of NUREG-1431 Specifications
(Continued)**

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES CONTENTS
ITS 1.0, "USE AND APPLICATIONS," has no Bases Section

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(3 Pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 1.0

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER

JUSTIFICATION

- 1.1-1 The NUREG-1431 definition of CHANNEL CALIBRATION states, "The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions." This change clarifies what encompasses the entire channel by rewording the definition to state, "The CHANNEL CALIBRATION shall encompass those components, such as sensors, alarms, displays, and trip functions required to perform the specified safety function(s)." The COT and TADOT definitions are also similarly revised. This change is consistent with TSTF-64.
- 1.1-2 Not Used.
- 1.1-3 Adds new example 1.4-4 to ITS 1.4 to clarify meaning of SR notes of the type, "Only required to be performed in MODE..." This change is consistent with WOG- 74, Rev. 1.
- 1.1-4 Not Used.
- 1.1-5 The definitions for ESF RESPONSE TIME and RTS RESPONSE TIME would be revised to substitute the word "verified" in lieu of "measured," consistent with the terminology of NUREG-1431, SR 3.3.1.16, and SR 3.3.2.10. This change would ensure consistency between the definitions for time and the requirements to periodically verify response time is within limits. This change is consistent with Industry Traveler TSTF-111, Rev. 1.
- 1.1-6 The definition of the PTLR would be revised to include the maximum allowable PORV lift settings and arming temperature associated with the [low temperature Over pressurization protection (LTOP)] system, and to be consistent with the CORE OPERATING LIMITS REPORT (COLR) definition. ITS 3.4.12 states that the PORV lift settings are specified in the PTLR. The current definition for PTLR does not identify these lift settings as being contained in the PTLR.
- The [LTOP] arming temperature was added to the PTLR since changes in the heatup/cooldown figures could change the arming temperature. This change corrects the PTLR definition to be consistent with all of the requirements contained in the PTLR. Referenced methodologies for the PTLR would contain the methodology used to develop the heatup and cooldown figures, as well as the methodology for developing the [LTOP] setpoints. This change is consistent with Industry Traveler WOG 67, Rev. 1.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 1.0

CHANGE

NUMBER

JUSTIFICATION

- 1.1-7 The definition for CFT in the current DCPD TS will be retained in the ITS. CFT is in active use in numerous procedures in the plant. The CFT is used in applications for which the COT is not fully suitable. Although CFT and COT definitions appear similar, there is one important difference. Strict adherence to COT requirements includes quantitative adjustments as appropriate to bring setpoints into the desired range. This requirement for quantitative adjustment cannot be satisfied in a reasonable manner on some components/sensors/channels due to their design. However, CFT is a qualitative test to determine functionality. A loss of function indicated by the CFT results in a notification to restore the functional performance, following existing procedures. The CFT definition is in the DCPD CTS. The words "or actual," "required," and the "CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested" are added to the definition of CFT consistent with NUREG-1431 definition for COT.
- 1.1-8 The reactor vessel head closure bolt requirements for MODES 4, 5, and 6 are clarified. Footnote b, is revised for MODES 4 and 5 to refer to "the required reactor vessel head closure bolts fully tensioned" and Note c for MODE 6 is revised to read "the required reactor head closure bolts less than fully tensioned." The transition point between MODES 5 and 6 would also be clarified as occurring when the required reactor vessel head closure bolts are less than fully tensioned. The required number of closure bolts, which may be less than the total number, is established by analysis that demonstrates adequate O-ring compression to prevent leakage and ensures the ASME Section III stress limits for affected components are not exceeded. This change is consistent with Industry Traveler TSTF-88.
- 1.1-9 The definitions of COT, [CFT], and TADOT are expanded to include the details of acceptable performance methodology. Performance of these tests in a series of sequential, overlapping, or total channel steps provides the necessary assurance of appropriate operation of the entire channel. This change also makes the COT and TADOT definitions consistent with the NUREG-1431 definition of CHANNEL CALIBRATION which already contains similar wording. This change is consistent with Industry Traveler TSTF-39, Rev. 1.
- 1.1-10 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 1.1-11 This change adds a new example (1.4-5) to ITS Section 1.4 to clarify surveillance Frequencies that are contingent on both a "specified frequency" and plant conditions. The ITS contains many surveillance Frequencies that are contingent on both a "specified frequency" and plant conditions. For example, "Within 7 days prior to the initiation of PHYSICS TESTS," and "Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days." These Frequencies do not fall clearly under any of the existing Section 1.4 examples. The proposed example is needed to make clear that: (1) the SR 3.0.2 extension of 1.25 times the specified Frequency applies to the specified Frequency, and (2) that the interval allowed to perform a missed surveillance by SR 3.0.3 applies.
- SR 3.0.2 is clear that the 1.25 extension may be applied to "the interval specified in the Frequency," so the proposed change does not change the intent of the specifications. SR 3.0.2 applies if a surveillance is not performed within the "specified Frequency."

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 1.0

CHANGE

NUMBER

JUSTIFICATION

1.1-11
(continued)

Again, the example does not change the intent of the specifications but only makes clear the application of SR 3.0.2 and 3.0.3 to surveillances with Frequencies tied to plant conditions. This change will eliminate confusion and misapplication of the ITS and will ensure consistent application of SR 3.0.2 and 3.0.3 to these types of surveillance Frequencies. This change is consistent with Industry Traveler WOG-90.

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(2 Pages)

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 1.0

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
1.1-1	This change would clarify what encompasses the entire channel by rewording the definition to state, "The CHANNEL CALIBRATION shall encompass those components, such as sensors, alarms, displays, and trip functions, required to perform the specified safety function(s)." The COT and TADOT definitions are similarly revised.	Yes	Yes	Yes	Yes
1.1-2	Not Used.	N/A	N/A	N/A	N/A
1.1-3	Not Used.	Yes	Yes	Yes	Yes
1.1-4	Not Used.	N/A	N/A	N/A	N/A
1.1-5	The definitions for ESF RESPONSE TIME and RTS RESPONSE TIME would be revised to substitute the word "verified" in lieu of "measured" consistent with the requirements of NUREG-1431, SR 3.3.1.16 and SR 3.3.2.10.	Yes	Yes	Yes	Yes
1.1-6	The definition of the PTLR would be revised to include the maximum allowable PORV lift settings and the arming temperature associated with the LTOP system, and to be consistent with the COLR definition.	Yes	Yes	Yes	Yes
1.1-7	The definition of CFT in the CTS will be retained in the ITS. NUREG-1431 does not include the definition of this test.	Yes	No, not part of CTS.	No, not part of CTS.	No, not part of CTS.
1.1-8	Note b, is revised to refer to the "required number of reactor vessel head closure bolts fully tensioned" and Note c is revised to read "Required reactor head closure bolts less than fully tensioned."	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 1.0

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
1.1-9	The definition of COT, [CFT], and TADOT are expanded to include the details of acceptable performance methodology. Performance of this test in a series of sequential, overlapping, or total channel steps provides the necessary assurance of appropriate operation of the entire channel.	Yes	Yes	Yes	Yes
1.1-10	This change is based on the CTS definition of CONTROLLED LEAKAGE. This change is a clarification only and does not affect the way RCS water inventory balances are performed.	No, not part of CTS.	No, not part of CTS.	No, maintaining ISTS wording.	Yes
1.1-11	This change adds a new example (1.4-5) to ITS Section 1.4 to clarify surveillance frequencies that are contingent on both specified frequency and plant conditions.	Yes	Yes	Yes	Yes

JLS Conversion to Improved Technical Specifications Diablo Canyon Power Plant

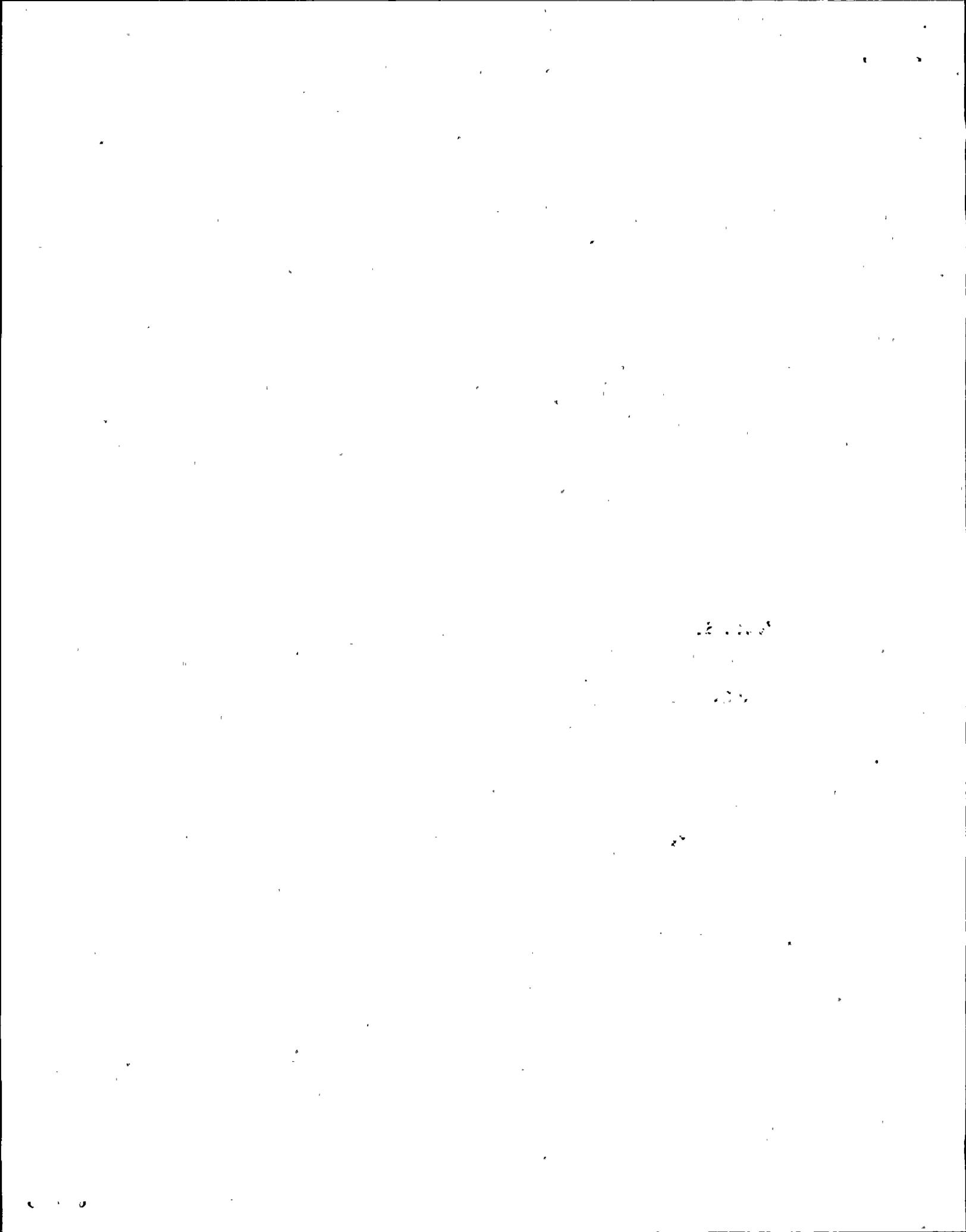
Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 2.0 - Safety Limits and Limiting Safety System Settings

ITS 2.0 - Safety Limits



11

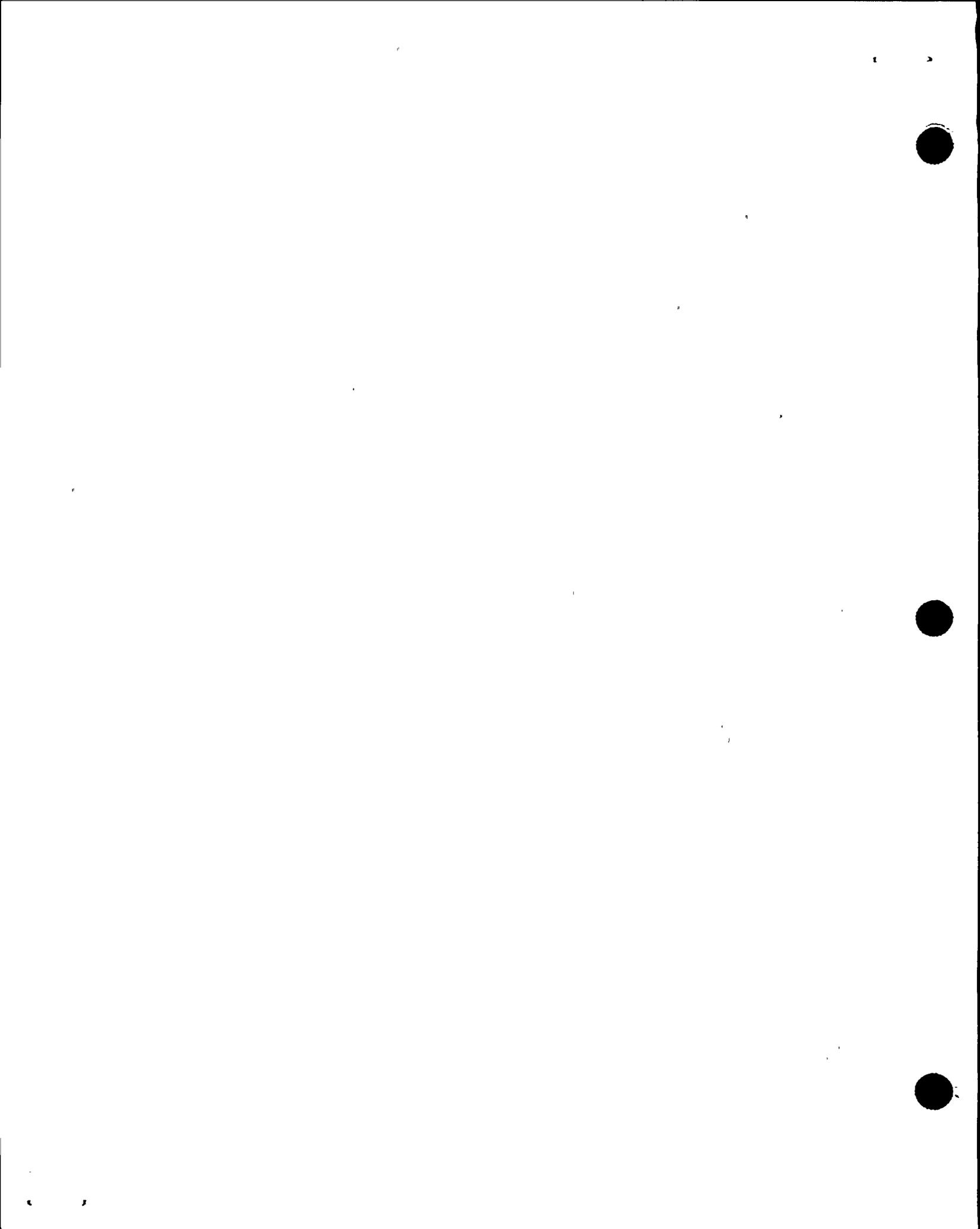


IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 2.0

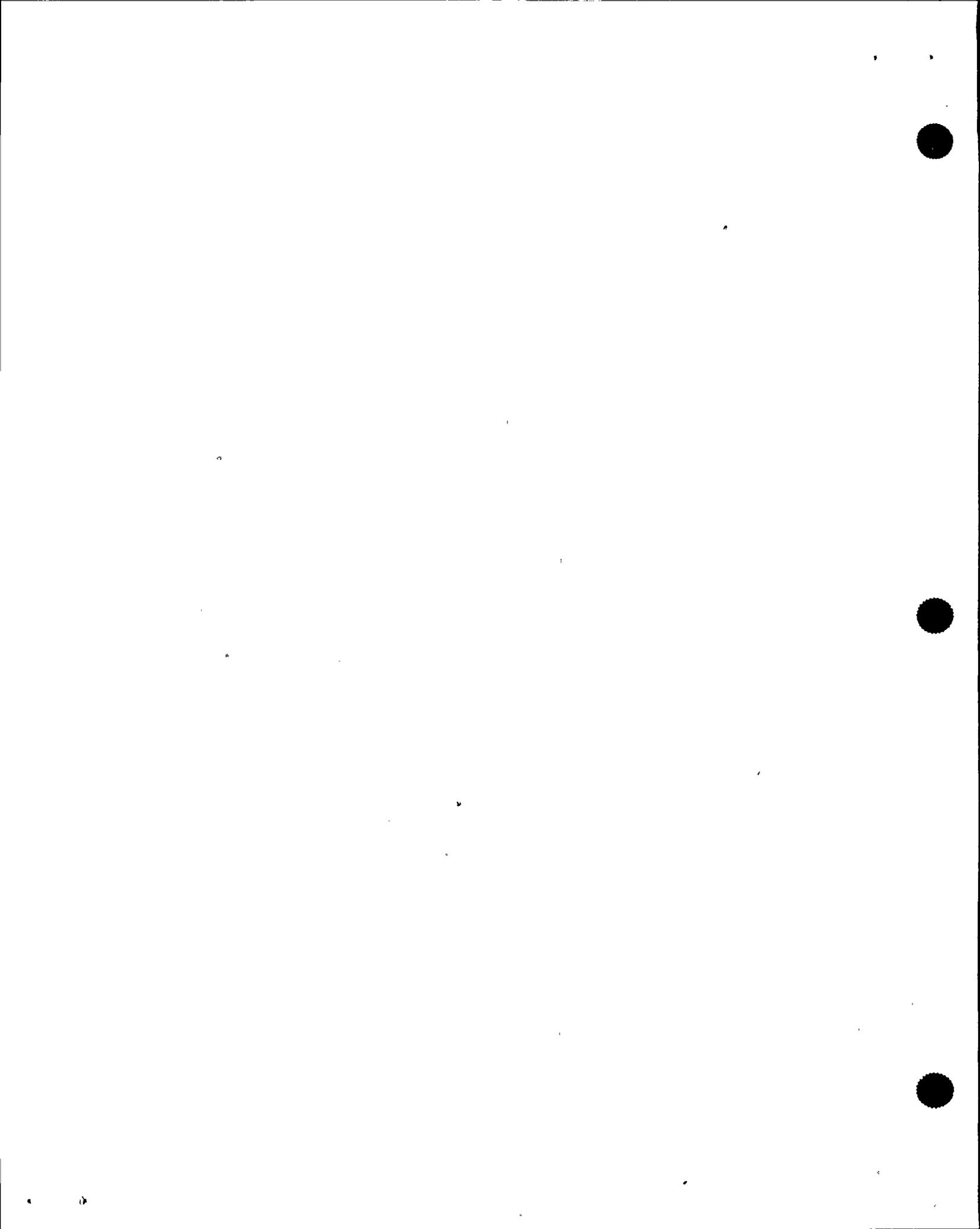
CONTENTS

- ENCLOSURE 1 - CROSS-REFERENCE TABLES
- ENCLOSURE 2 - MARK-UP OF CURRENT TS
- ENCLOSURE 3A - DESCRIPTION OF CHANGES TO CURRENT TS
- ENCLOSURE 3B - CONVERSION COMPARISON TABLE - CURRENT TS
- ENCLOSURE 4 - NO SIGNIFICANT HAZARDS CONSIDERATIONS
- ENCLOSURE 5A - MARK-UP OF NUREG-1431 SPECIFICATIONS
- ENCLOSURE 5B - MARK-UP OF NUREG-1431 BASES
- ENCLOSURE 6A - DIFFERENCES FROM NUREG-1431
- ENCLOSURE 6B - CONVERSION COMPARISON TABLE - NUREG 1431



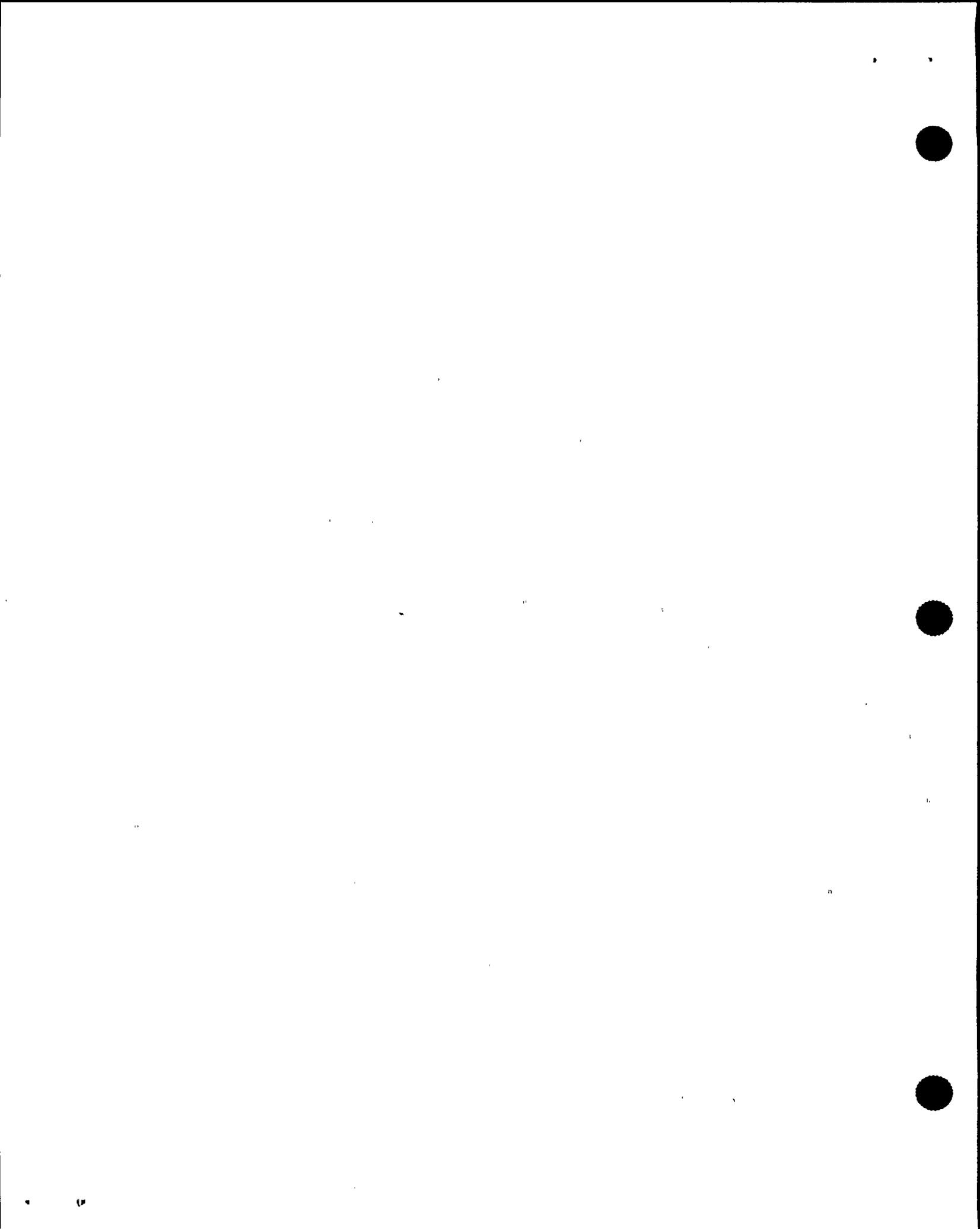
ENCLOSURE 1

CROSS-REFERENCE TABLES



CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(1 Page)
CONVERSION TABLE SORTED BY IMPROVED TS	(1 Page)
METHODOLOGY	(3 Pages)



CROSS-REFERENCE TABLE FOR 2.0
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
2.1.1	SL			2.1.1	SL		
2.1.1	ACTION		01-02- A	2.2.1	VIOLATION		
2.1.1	FIG	2.1-1		2.1.1	FIG	2.1.1-1	
2.1.2	SL			2.1.2	SL		
2.1.2	ACTION	MODES 1, 2	01-02- A	2.2.2.1	VIOLATION	MODES 1, 2	
2.1.2	ACTION	MODES 3, 4, 5	01-02-A	2.2.2.2	VIOLATION	MODES 3, 4, 5	
2.2.1	LSSS		02-01-A	3.3.1	Table	3.3.1-1	
2.2.1	ACTION	a	02-06-LG	3.3.1	Table	3.3.1-1	
2.2.1	ACTION	b	02-06-LG	3.3.1	Table	3.3.1-1	
2.2.1	TABLE	2.2-1		3.3.1	Table	3.3.1-1	



CROSS-REFERENCE TABLE FOR 2.0
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
2.1.1				2.1.1			
2.1.2				2.1.2			
Figure 2.1.1a				Figure 2.1.1-1a			
Figure 2.1.1b				Figure 2.1.1-1b			
2.1.1	ACTION		01-02-A	2.2.1			
2.1.2	ACTION	MODES 1 & 2	01-02-A	2.2.2.1			
2.1.2	ACTION	MODES 3-5	01-02-A	2.2.2.2			
		NA		2.2.3		Not Used	
		NA		2.2.4		Not Used	
		NA		2.2.5		Not Used	
		NA		2.2.6		Not Used	



Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

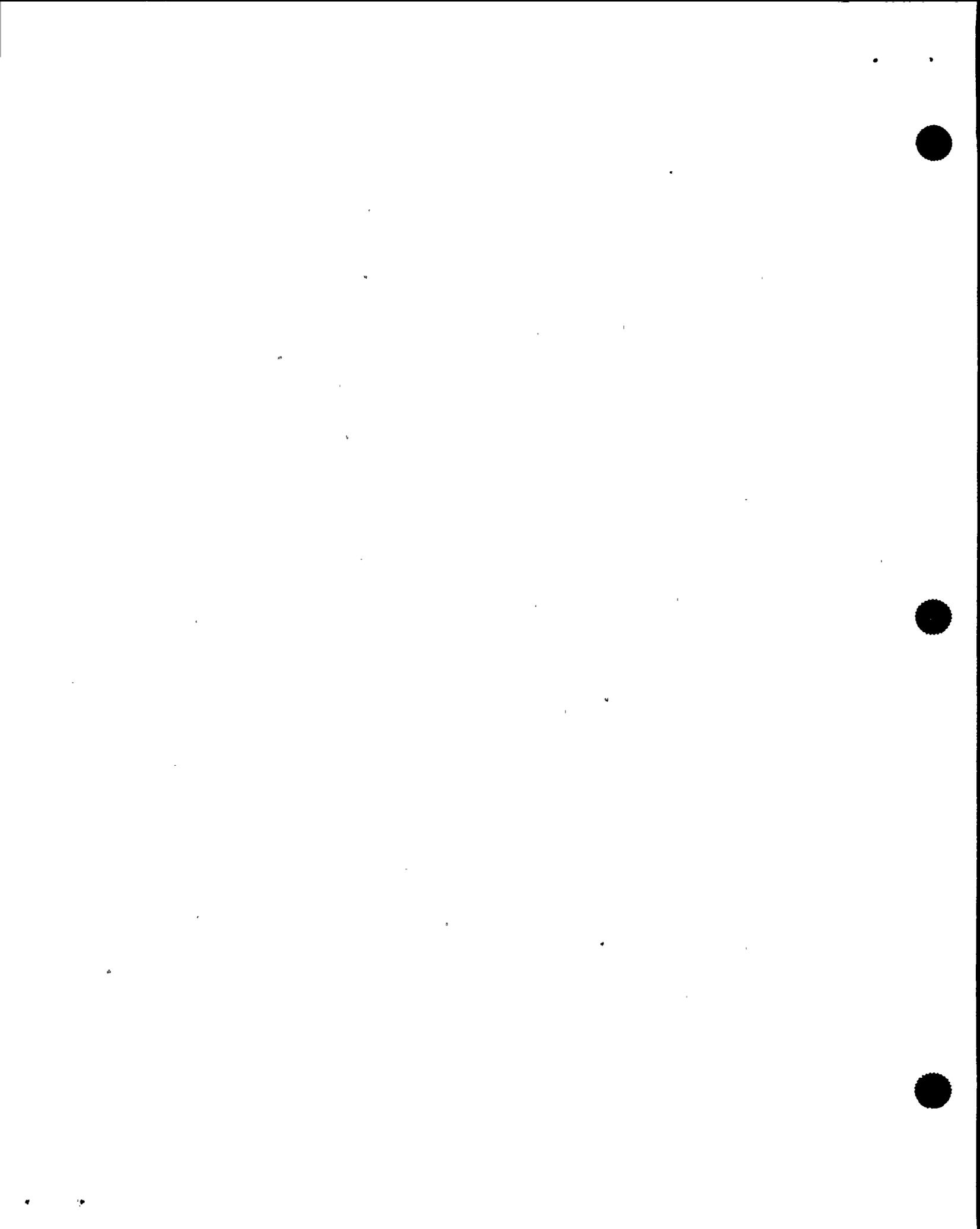
Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.



Methodology for Cross-Reference Tables
(Continued)

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated This item is relocated to another licensee control document outside the TS (see Code for specific reference location).

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

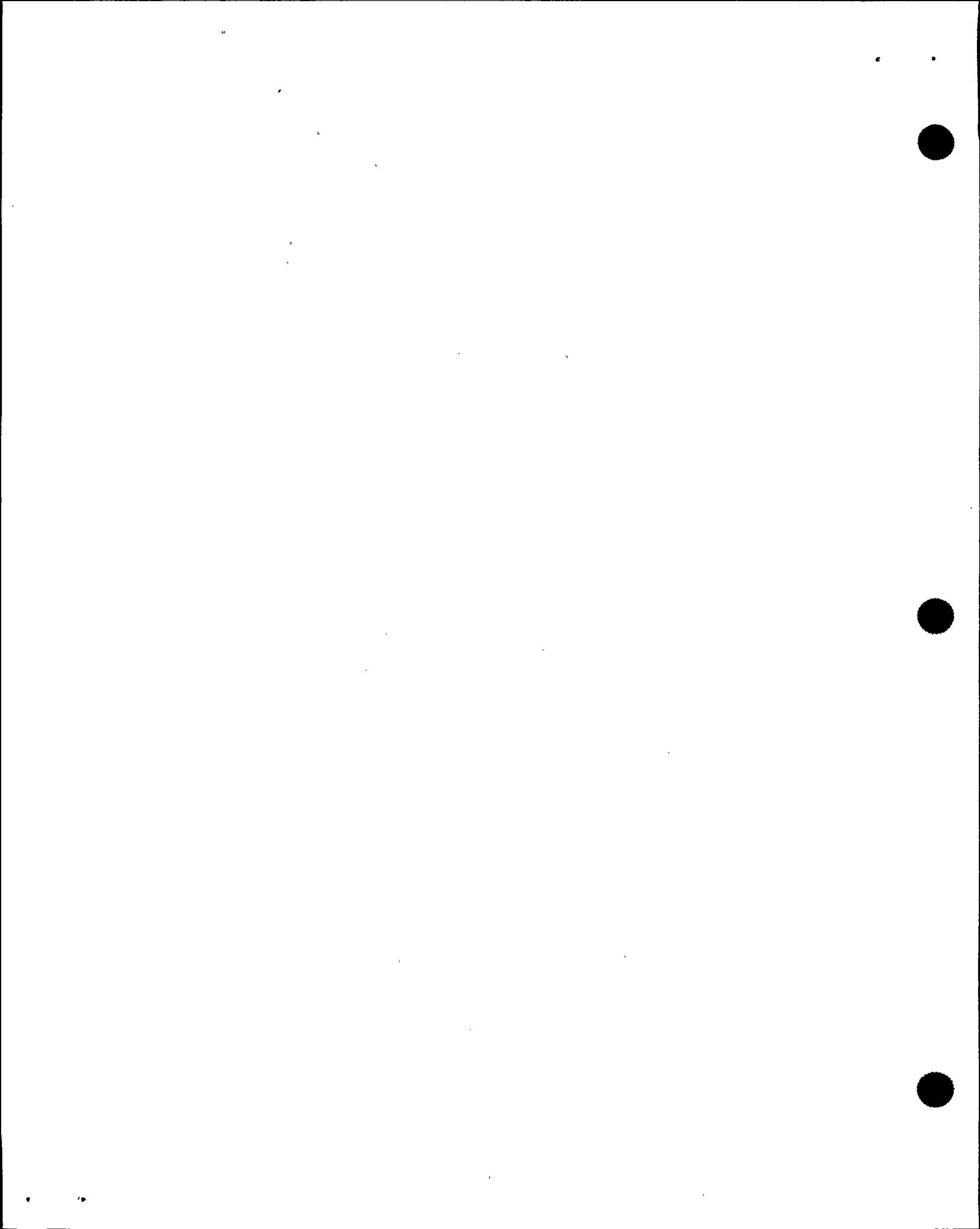
In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

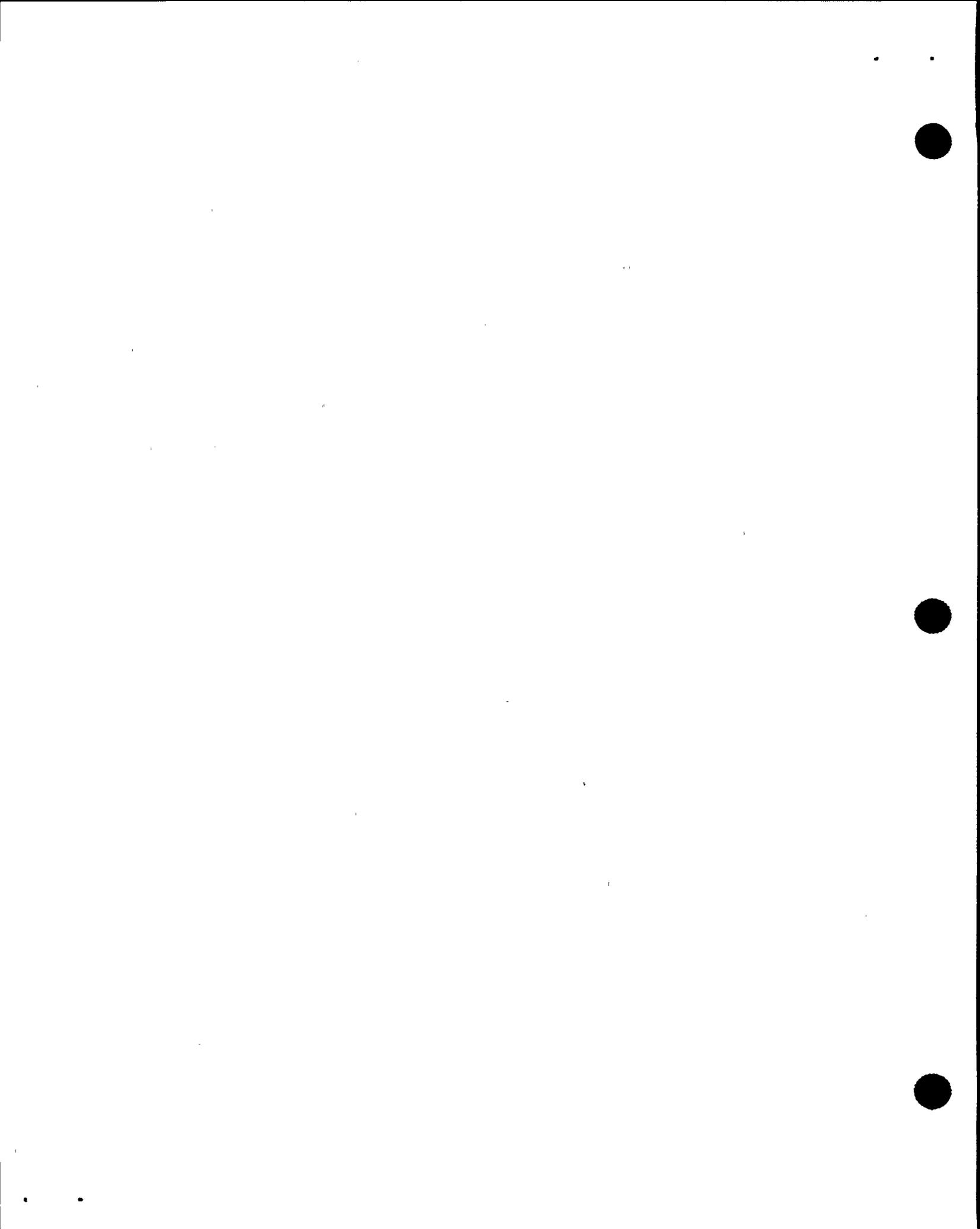
New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).
Note:	The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."



Methodology for Cross-Reference Tables
(Continued)

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

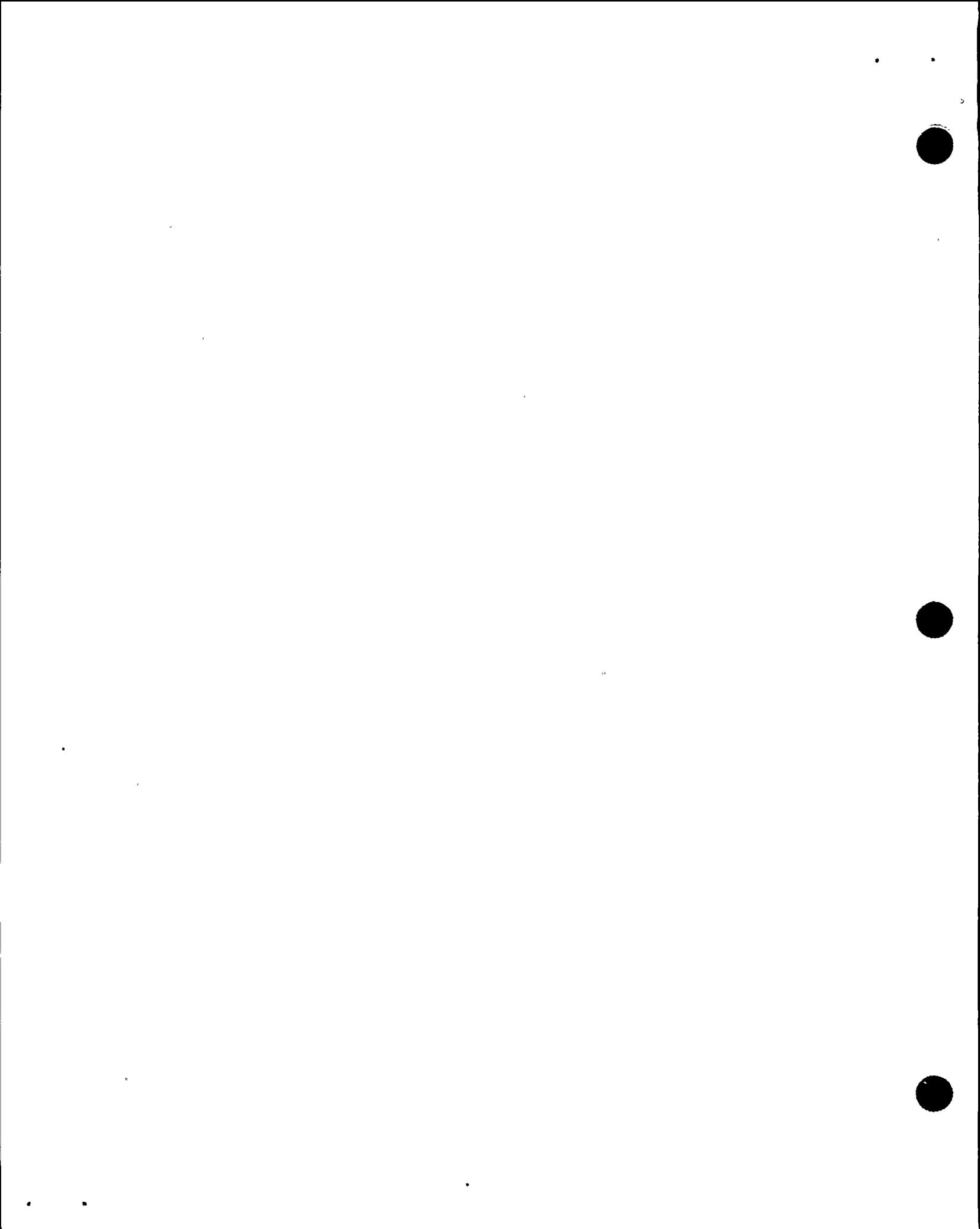


ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
2.1.1	2-1
2.1.2	2-1
2.2.1	2-3
Methodology	(2 Pages)



2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour ~~and comply with the requirements of Specification 6.7.~~

01-02-A

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, ~~and comply with the requirements of Specification 6.7.~~

01-02-A

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, ~~and comply with the requirements of Specification 6.7.~~

01-02-A



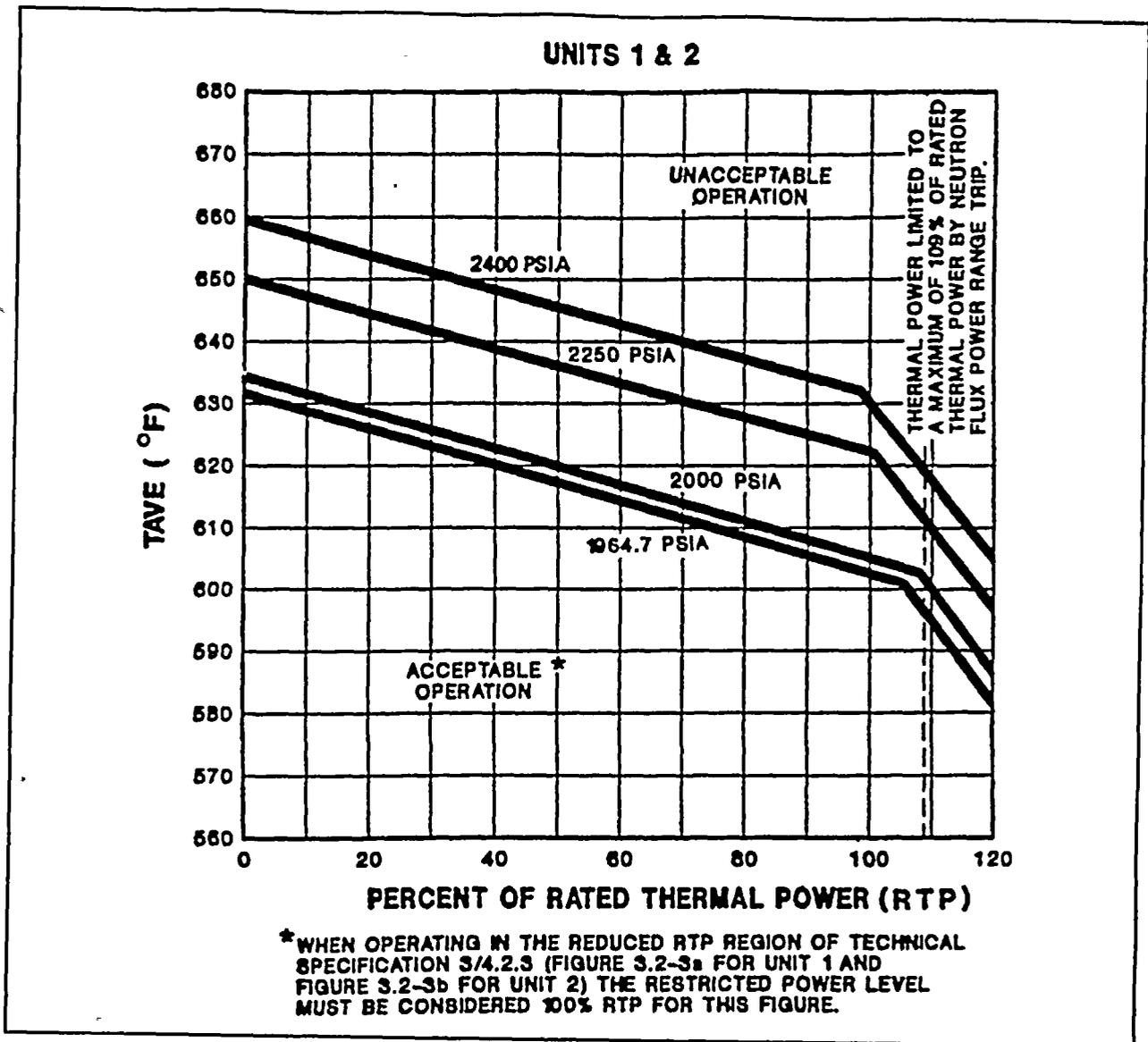
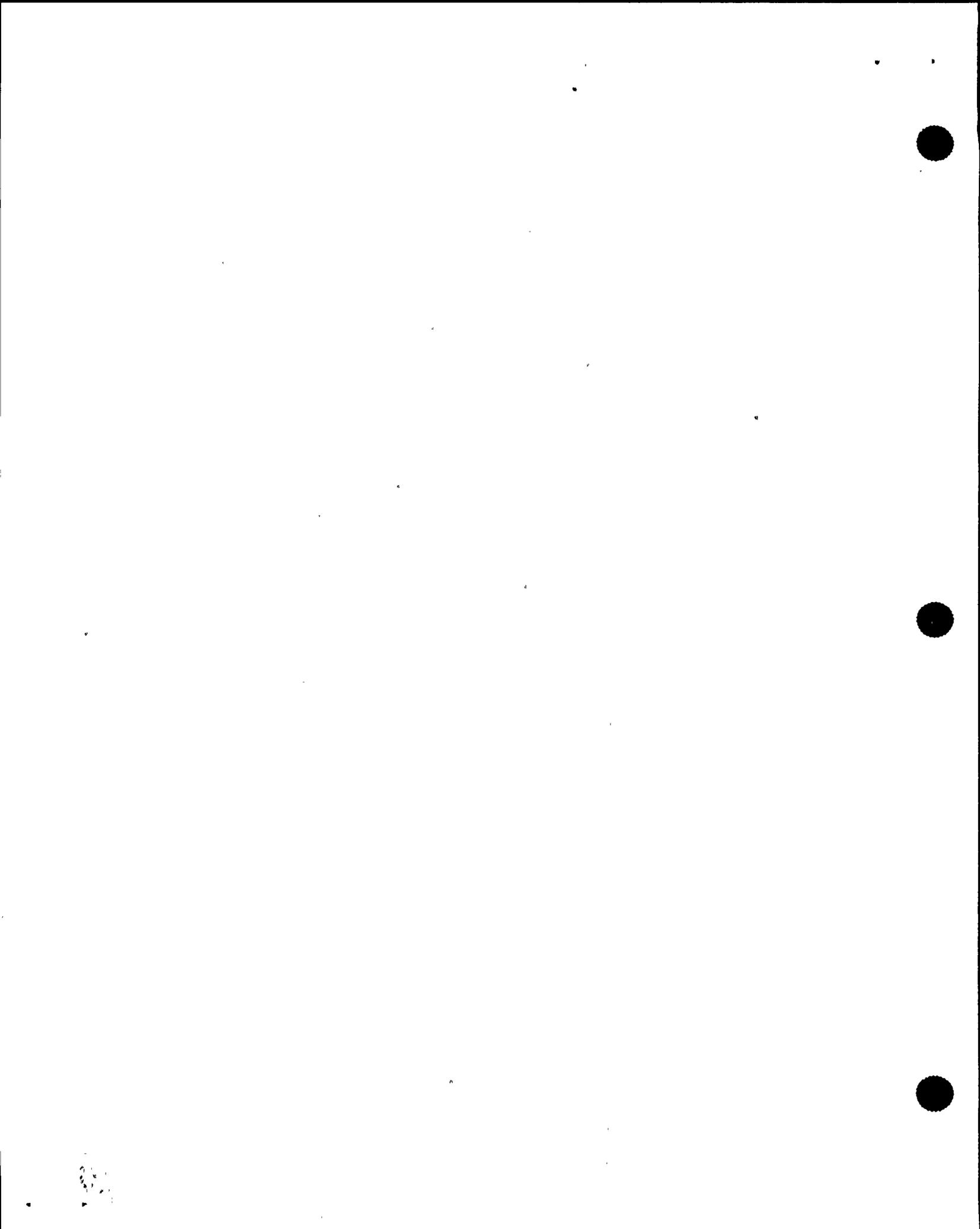


FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT



SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

02-01-A

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

~~2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.~~

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. ~~With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Values column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.~~

02-06-LG

b. ~~With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirements of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.~~

02-06-LG

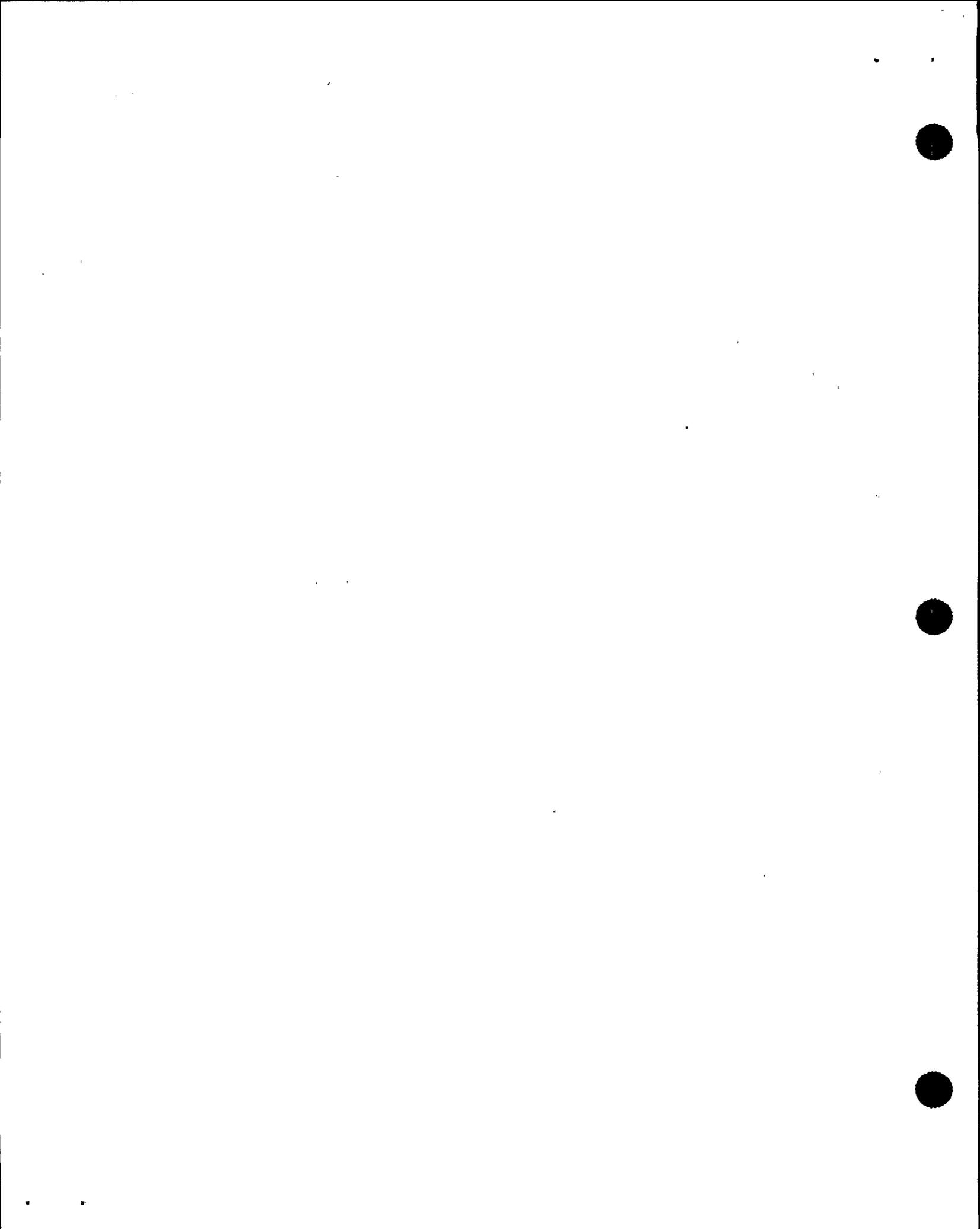


TABLE 2.2-1

02-01-A

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux a. Low Setpoint b. High Setpoint	\leq 25% of RATED THERMAL POWER \leq 109% of RATED THERMAL POWER	\leq 27.1% of RATED THERMAL POWER \leq 111.1% of RATED THERMAL POWER
3. Power Range, Neutron Flux High Positive Rate	\leq 5% of RATED THERMAL POWER with a time constant \geq 2 seconds	\leq 6.5% of RATED THERMAL POWER with a time constant \geq 2 seconds
4. Power Range, Neutron Flux High Negative Rate	\leq 5% of RATED THERMAL POWER with a time constant \geq 2 seconds	\leq 6.5% of RATED THERMAL POWER with a time constant \geq 2 seconds
5. Intermediate Range, Neutron Flux	\leq 25% of RATED THERMAL POWER	\leq 30.9% of RATED THERMAL POWER
6. Source Range, Neutron Flux	\leq 10^5 counts per second	\leq 1.4×10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 2
8. Overpower ΔT	See Note 3	See Note 4
9. Pressurizer Pressure-Low	\geq 1950 psig	\geq 1944.4 psig
10. Pressurizer Pressure-High	\leq 2385 psig	\leq 2390.6 psig
11. Pressurizer Water Level-High	\leq 92% of instrument span	\leq 92.5% of instrument span
12. Reactor Coolant Flow-Low	\geq 90% of minimum measured flow** per loop	\geq 89.7% of minimum measured flow** per loop

**Minimum measured flow is 89,800 gpm per loop for Unit 1 and 90,625 gpm per loop for Unit 2.



TABLE 2.2-1 (Continued)

02-01-A

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level-Low-Low	≥ 7.2% of narrow range instrument span-each steam generator	≥ 6.8% of narrow range instrument span-each steam generator
Coincident with:		
a. RCS Loop ΔT Equivalent to Power ≤50% RTP	RCS Loop ΔT variable input ≤50% RTP	RCS Loop ΔT variable input ≤51.5% RTP
With a time delay (TD)		
Or		
b. RCS Loop ΔT Equivalent to Power >50% RTP	RCS Loop ΔT variable input >50% RTP	RCS Loop ΔT variable input >51.5% RTP
With no time delay		
14. DELETED		
15. Undervoltage-Reactor Coolant Pumps	≥ 8050 volts-each bus	≥ 7730 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 54.0 Hz - each bus	≥ 53.9 Hz - each bus
17. Turbine Trip		
a. Low Autostop Oil Pressure	≥ 50 psig	≥ 45 psig
b. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open
18. Safety Injection Input from ESF	N.A.	N.A.
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.
20. Reactor Trip Breakers	N.A.	N.A.
21. Automatic Trip and Interlock Logic	N.A.	N.A.



TABLE 2.2-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

02-01-A

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
22. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amps	$\geq 6 \times 10^{-11}$ amps
b. Low Power Reactor Trips Block, P-7		
1) P-10 Input	10% of RATED THERMAL POWER	> 7.9%, < 12.1% of RATED THERMAL POWER
2) P-13 Input Pressure Equivalent	< 10% RTP Turbine Impulse Pressure Equivalent	$\leq 12.1\%$ RTP Turbine Impulse
c. Power Range Neutron Flux, P-8	< 35% of RATED THERMAL POWER	< 37.1% of RATED THERMAL POWER
d. Power Range Neutron Flux, P-9	< 50% of RATED THERMAL POWER	< 52.1% of RATED THERMAL POWER
e. Power Range Neutron Flux, P-10	10% of RATED THERMAL POWER	> 7.9%, < 12.1% of RATED THERMAL POWER
f. Turbine Impulse Chamber Pressure, P-13	< 10% RTP Turbine Impulse Pressure Equivalent	< 12.1% RTP Turbine Impulse Pressure Equivalent
23. Seismic Trip	≤ 0.35 g	≤ 0.40 g

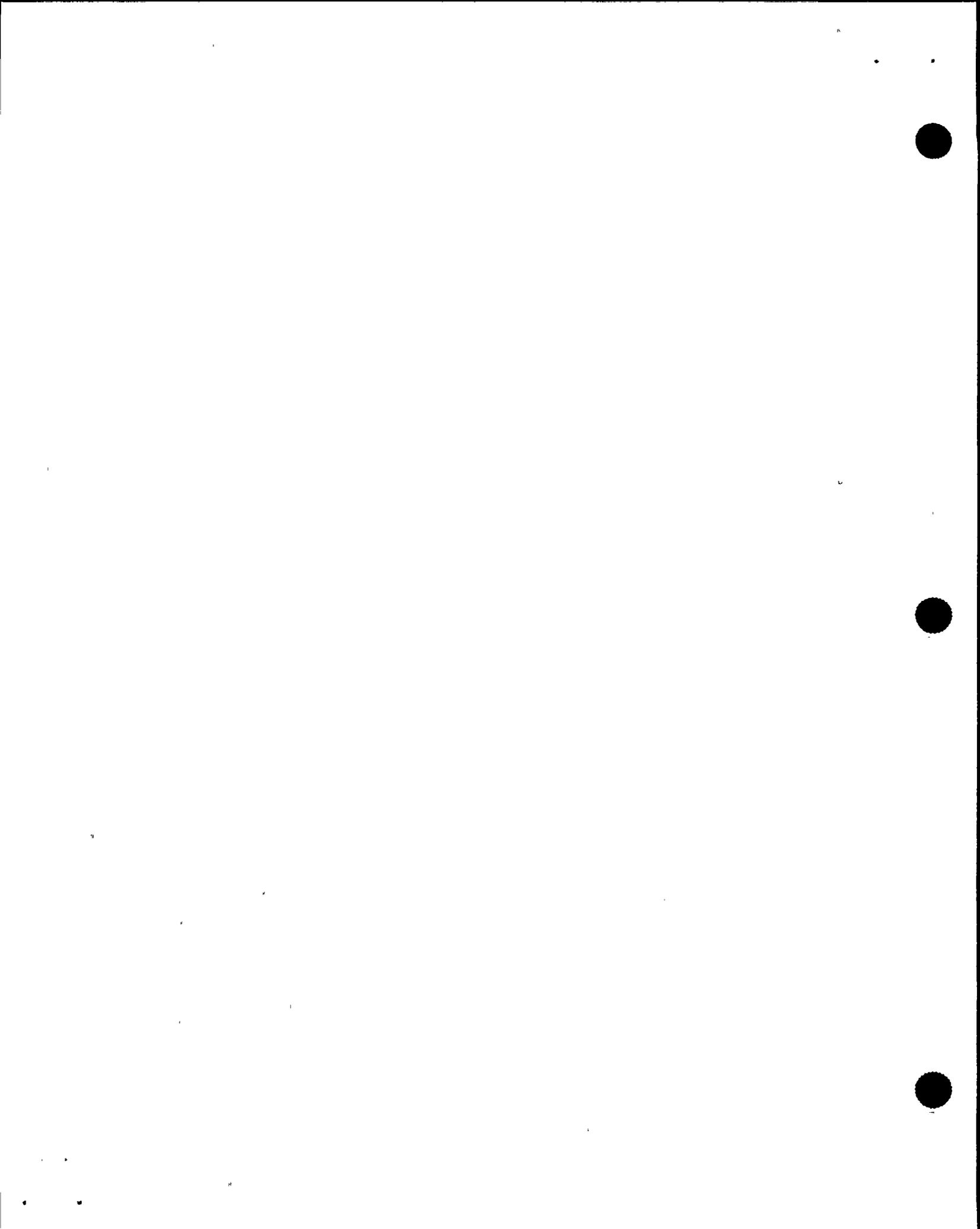


TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

02-01-A

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \left(\frac{1+\tau_4 S}{1+\tau_5 S} \right) \leq \Delta T_0 \left\{ K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S} \right) [T - T'] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: $\frac{1+\tau_4 S}{1+\tau_5 S}$ = Lead-lag compensator on measured ΔT

τ_4, τ_5 = Time constants utilized in the lead-lag controller for ΔT , $\tau_4 = 0$ seconds,
 $\tau_5 = 0$ seconds

ΔT_0 = Indicated ΔT at RATED THERMAL POWER

$K_1 = 1.2$

$K_2 = 0.0182/^\circ F$

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg}
 dynamic compensation

τ_1, τ_2 = Time constants utilized in the lead-lag controller for T_{avg} , $\tau_1 = 30$ seconds
 $\tau_2 = 4$ seconds

T = Average temperature, $^\circ F$;

02-05-A

02-04-M

02-04-M

02-04-M



TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

02-01-A

TABLE NOTATIONS

NOTE 1: (Continued)

T' = Nominal T_{avg} at RATED THERMAL POWER ~~≈ 576.6 °F (Unit 1) and ≈ 577.6 °F (Unit 2)~~

02-07-A

K_3 = 0.000831/psig

P = Pressurizer pressure, psig

P' = 2235 psig (Nominal RCS operating pressure)

S = Laplace transform operator, S^{-1}

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

- (I) for $q_t - q_b$ between - 19% and + 7%, $f_1(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).
- (ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds - 19%, the ΔT Trip Setpoint shall be automatically reduced by 2.75% of its value at RATED THERMAL POWER.
- (iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds + 7%, the ΔT Trip Setpoint shall be automatically reduced by 2.38% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.0% ΔT span.



TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

02-01-A

TABLE NOTATIONS

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{1+\tau_4 S}{1+\tau_5 S} \leq \Delta t_o \{ K_4 - K_5 \frac{\tau_3 S}{1+\tau_3 S} \} T - K_6 [T - T''] - f_2(\Delta I)$$

Where: $\frac{1+\tau_4 S}{1+\tau_5 S}$ = Lead-lag compensator on measured ΔT

τ_4, τ_5 = Time constants utilized in the lead-lag controller for ΔT , $\tau_4 = 0$ seconds, $\tau_5 = 0$ seconds

02-04-M

Δt_o = Indicated ΔT at RATED THERMAL POWER

$K_4 = 1.072$

02-04-M

$K_5 = 0.0174/^\circ F$ for increasing average temperature, and 0 for decreasing average temperature

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate-lag controller for T_{avg} dynamic compensation

τ_3 = Time constants utilized in the rate-lag controller for T_{avg} , $\tau_3 = 10$ secs.

$K_6 = 0.00145/^\circ F$ for $T > T''$, and 0 for $T \leq T''$

T = Average temperature, $^\circ F$

02-04-M

T'' = Indicated T_{avg} at RATED THERMAL POWER, $\leq 576.6^\circ F$ (Unit 1) and $\leq 577.6^\circ F$ (Unit 2)

02-07-A

S = Laplace transform operator, s^{-1}

$f_2(\Delta I) = 0$ for all ΔI



TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.0% ΔT span

NOTE 5: Steam Generator Water Level Low-Low Trip Time Delay

02-01-A

$$TD = B1(P)^3 + B2(P)^2 + B3(P) + B4$$

Where: P = RCS Loop ΔT Equivalent to Power (%RTP), $P \leq 50\%$ RTP

TD = Time delay for Steam Generator Water Level Low-Low Reactor Trip (in seconds).

$$B1 = -0.007128$$

$$B2 = +0.8099$$

$$B3 = -31.40$$

$$B4 = +464.1$$



Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. **Deletions** - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. **Additions** - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. **Modifications** - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. **Administrative** - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions** - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions** - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications** - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative** - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.



Methodology For Mark-Up of Current TS
(Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.



ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers

(1 Page)

Description of Changes

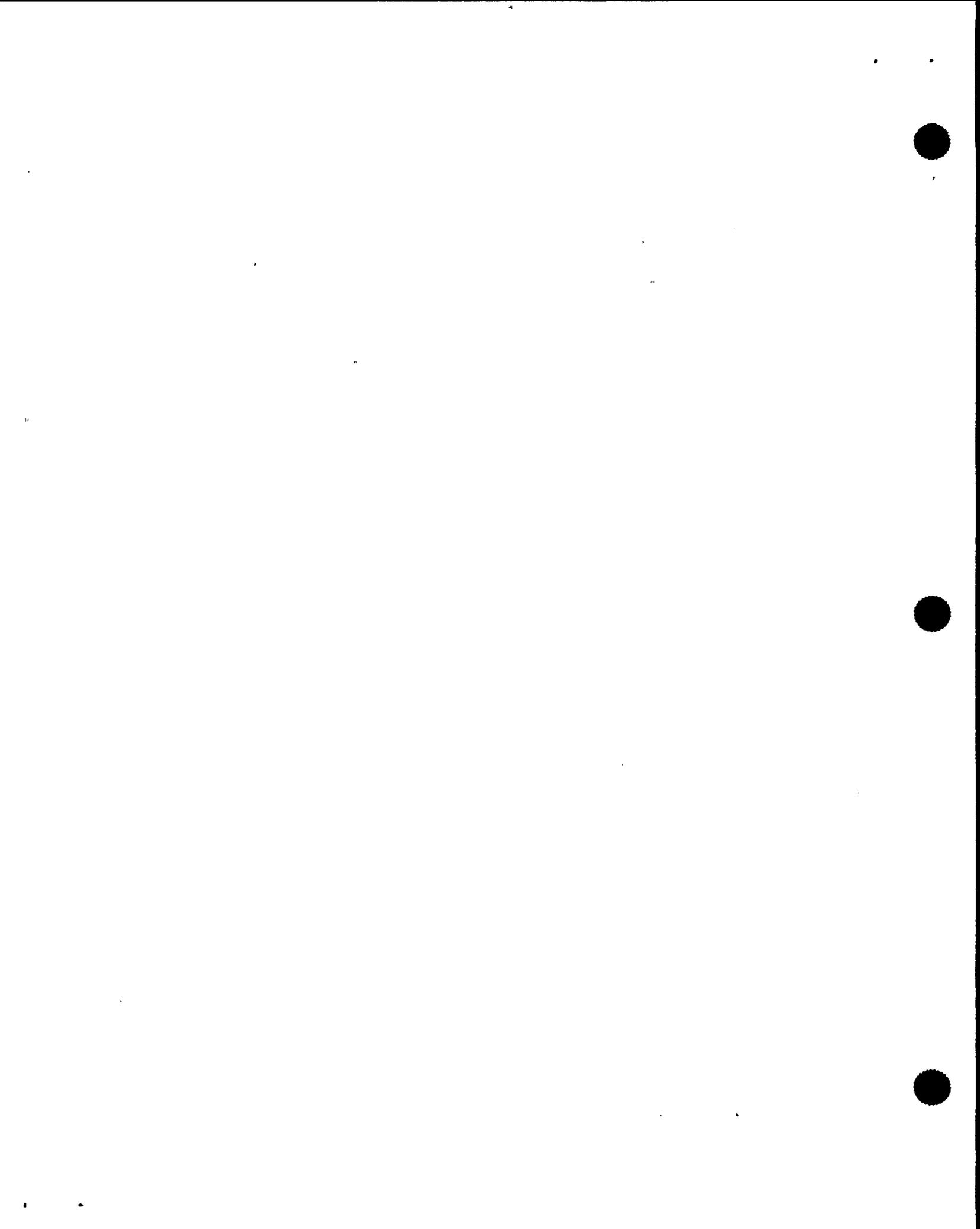
(1 Page)



TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS

SECTION 2.0

TECHNICAL SPECIFICATION TITLE	CHG. NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Safety Limits - Reactor Core	1	2.1.1	2.1.1	2.1.1	2.1.1
Safety Limits - Reactor Coolant System Pressure	1	2.1.2	2.1.2	2.1.2	2.1.2
Limiting Safety System Settings - Reactor Trip System Instrumentation Setpoints	2	2.2.1	2.2.1	2.2.1	2.2.1



DESCRIPTION OF CHANGES TO TS SECTION 2.0

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	A	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-02	A	The requirements embodied in separate administrative controls dealing with Safety Limits (SL) violations are deleted. Specifications 6.7 is deleted per CN 02-02-LS in Enclosure 3A of the Administrative Controls package.
02-01	A	The requirements of this LCO are moved to ITS LCO 3.3.1.
02-02	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-03	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-04	M	Addition of the inequality signs, consistent with NUREG-1431, indicates the conservative direction for these K values.
02-05	A	This change corrects the $OT_{\Delta T}$ equation by relocating the bracket to the correct position, as described in CN 3.3-10 of the CTS 3/4.3 attachment.
02-06	LG	The requirements stipulated in ACTIONS [a and b] are moved to ITS Table 3.3.1-1, with explicit direction contained in [the ITS Background (Trip Setpoints and Allowable Values) Bases, ACTIONS Bases, and SR 3.3.1.10 and SR 3.3.1.11 Bases]. This change removes details more appropriately controlled outside of the TS while retaining those aspects necessary to assure the protection functions are performed if necessary.
02-07	A	This change incorporates the values for T' (Nominal T_{avg} at RATED THERMAL POWER), that were inadvertently deleted during a previous License Amendment.



ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(1 page)



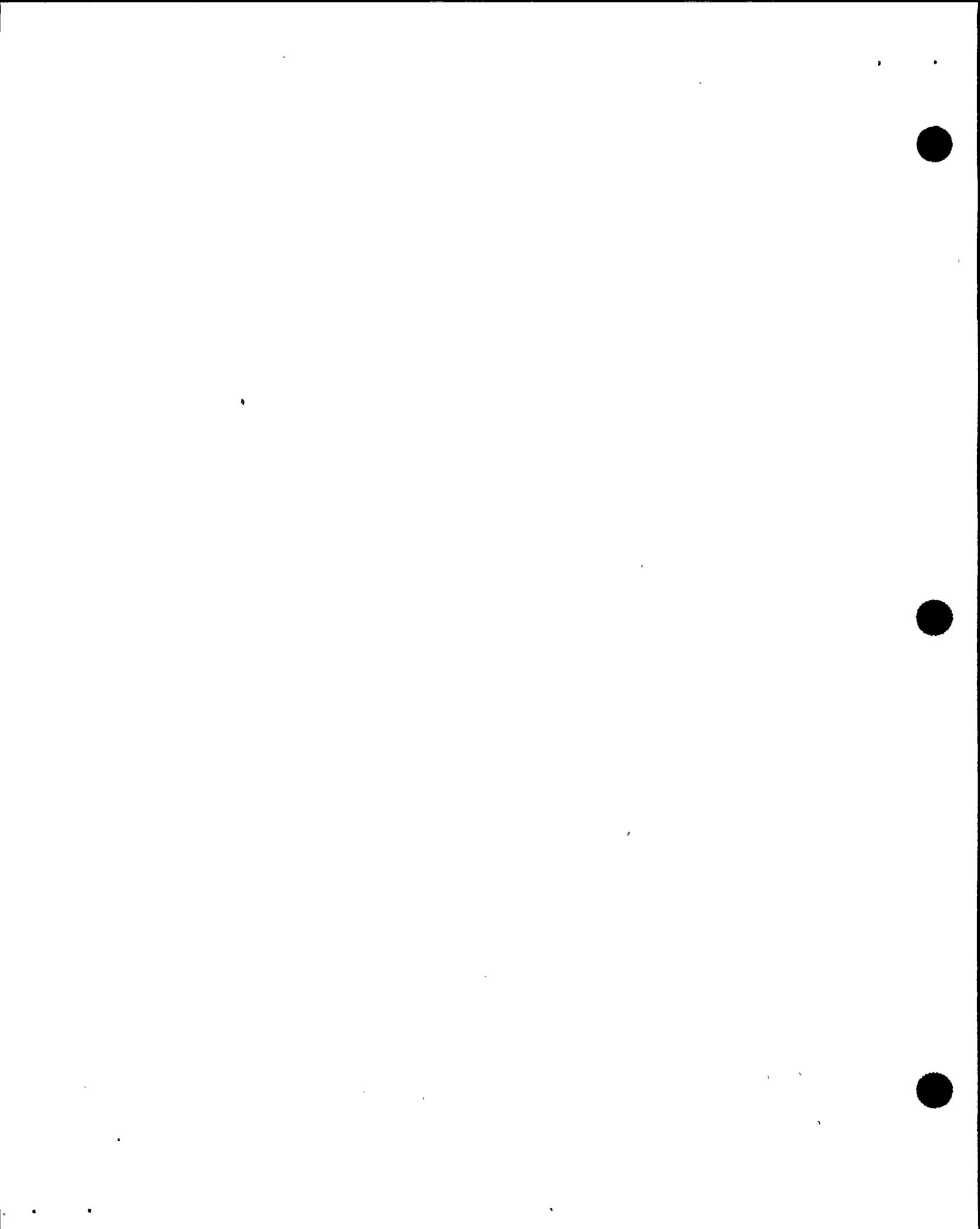
CONVERSION COMPARISON TABLE - CURRENT TS 2.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01-A	The specific restrictions on number of loops and licensed reactor power for power operation would be removed from the SLs. These restrictions are stated in other requirements of the license.	No, not in CTS.	No, not in CTS.	Yes	Yes
01-02-A	The requirements embodied in separate administrative controls dealing with SL violations are deleted.	Yes	Yes	Yes	Yes
02-01-A	The requirements of this LCO are moved to ITS LCO 3.3.1.	Yes	Yes	Yes	Yes
02-02-LG	The Reactor Trip System Instrumentation Trip Setpoints are moved to a licensee controlled document.	No, retained CTS. {See CN 02-01-A}	Yes, moved to ITS 3.3.1 Bases.	Yes, moved to USAR.	Yes, moved to ITS 3.3.1 Bases.
02-03-A	CTS ACTION b.1, Equation 2.2-1, and the values for Total Allowance (TA), Z, and Sensor Error (S) are deleted, consistent with NUREG-1431, as they are no longer required.	No, not in CTS.	No, not in CTS.	Yes	Yes
02-04-M	Addition of the inequality signs, {to the time constants and} to K_1 , K_4 , K_5 , and K_6 consistent with NUREG-1431, indicates the conservative direction for these K values {and τ values.}	Yes	No, retained CTS.	Yes	Yes
02-05-A	This change corrects the $OT\Delta T$ { } equation in the DCPD CTS by relocating the bracket to the correct position, as described in CN 3.3-10 of the {CTS 3/4.3} attachment.	Yes	No	No	No
02-06-LG	The requirements stipulated in ACTIONS [a and b] are moved to ITS Table 3.3.1-1, with direction contained in the ITS ACTION Bases.	Yes	Yes	Yes	Yes
02-07-A	This change incorporates the values for T' (Nominal T_{avg} at RATED THERMAL POWER), that were inadvertently deleted during a previous License Amendment for DCPD.	Yes	No	No	No



ENCLOSURE 4

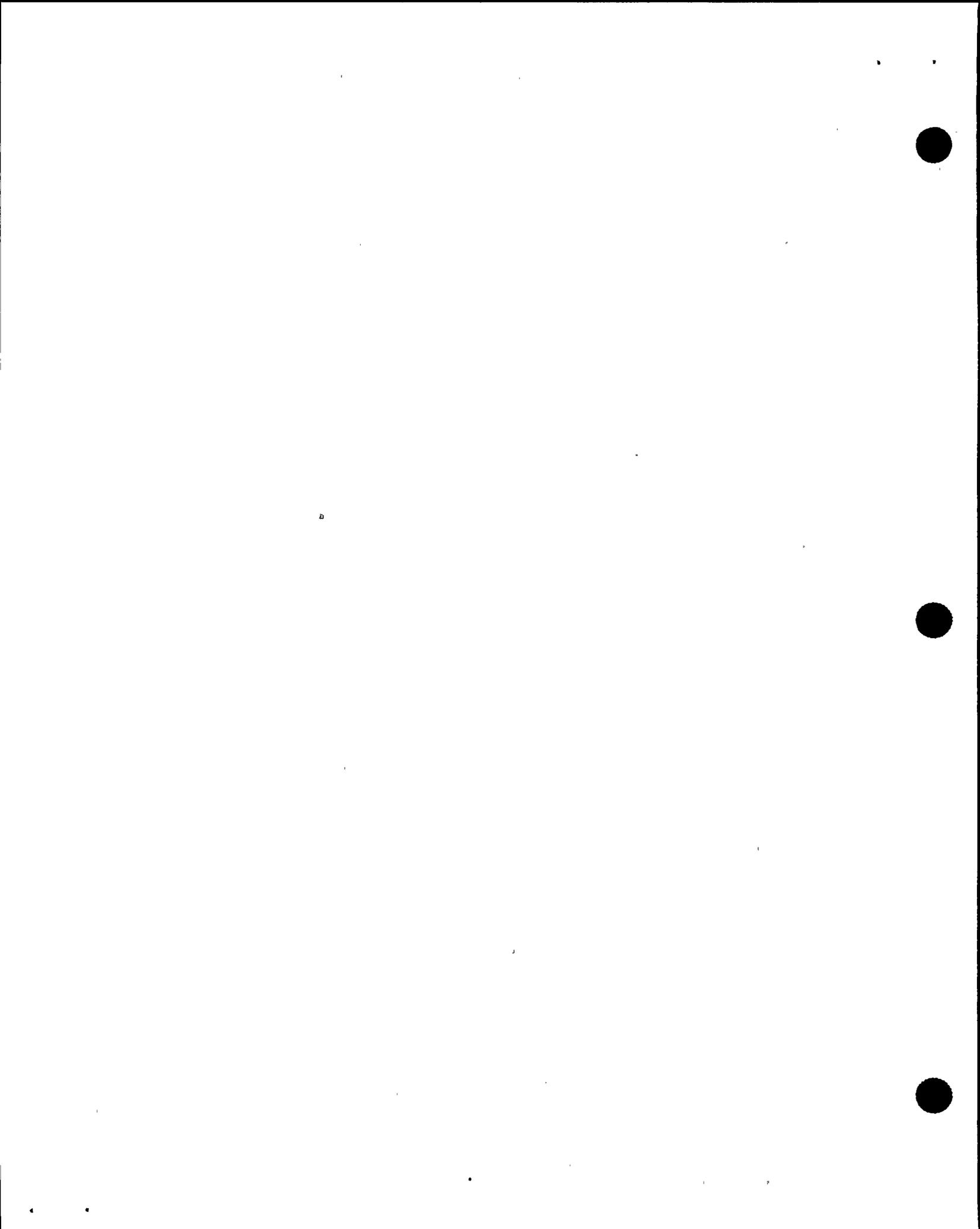
NO SIGNIFICANT HAZARDS CONSIDERATIONS



NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

PAGE

I.	Organization	2
II.	Description of NSHC Evaluations.	3
III.	Generic NSHCs	
	"A" - Administrative Changes	5
	"R" - Relocated Technical Specifications	7
	"LG" - Less Restrictive (moving information out of the TS)	10
	"M" - More Restrictive	12
IV.	Specific NSHCs - "LS"	
	None	
V.	Recurring NSHCs	
	None	



I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.



II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

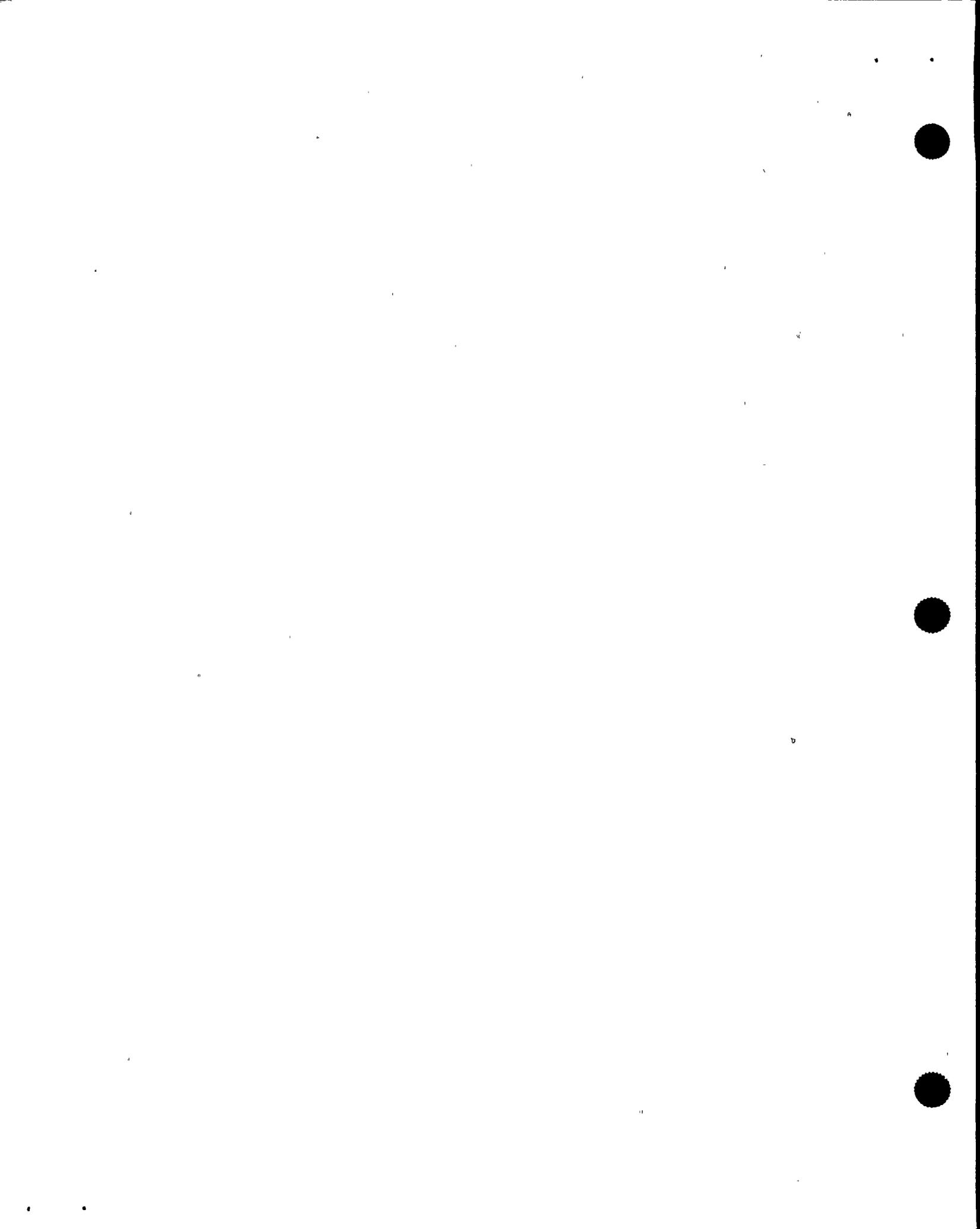
Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.



II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3..." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.



III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.



III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.



III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.



III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

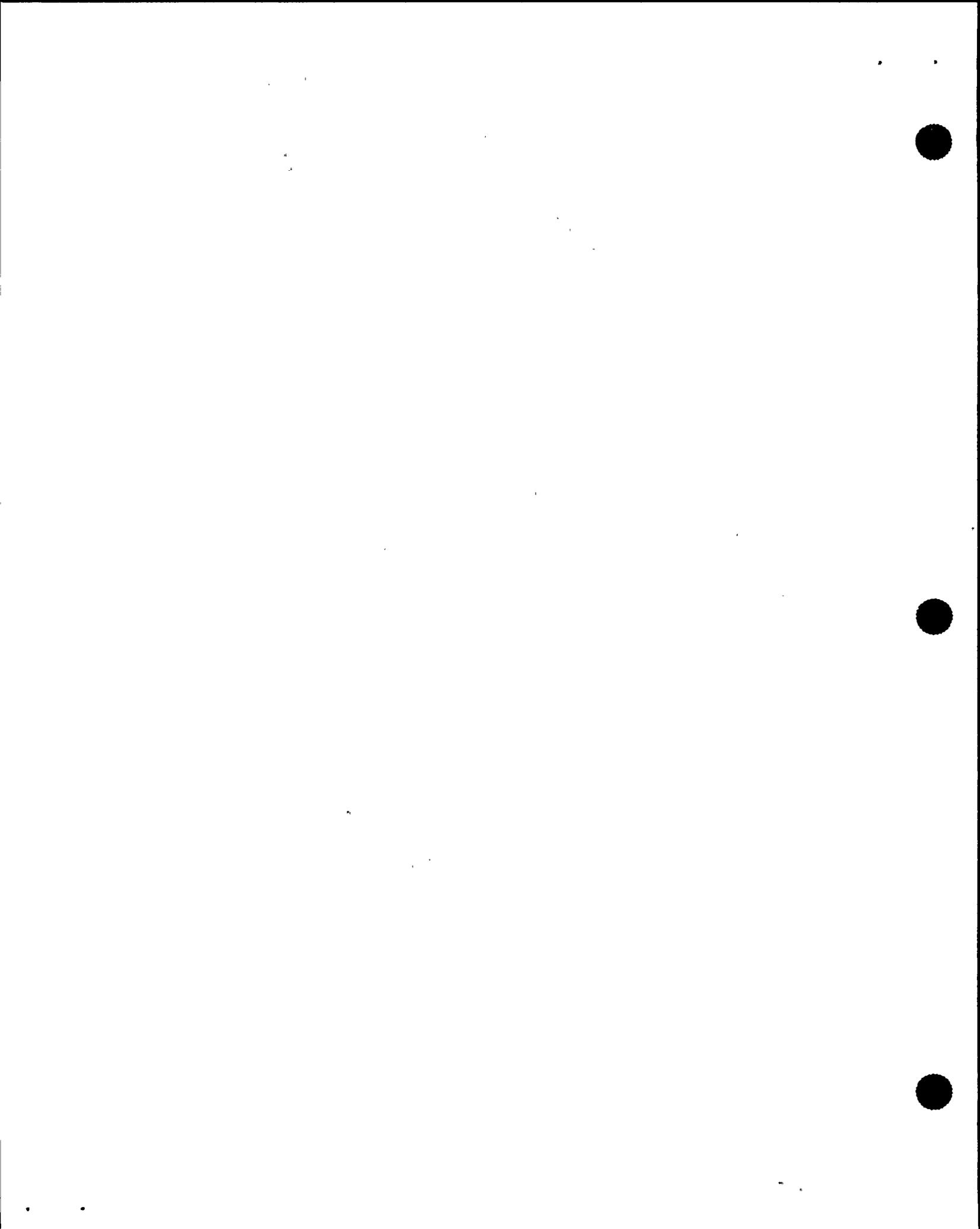
The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

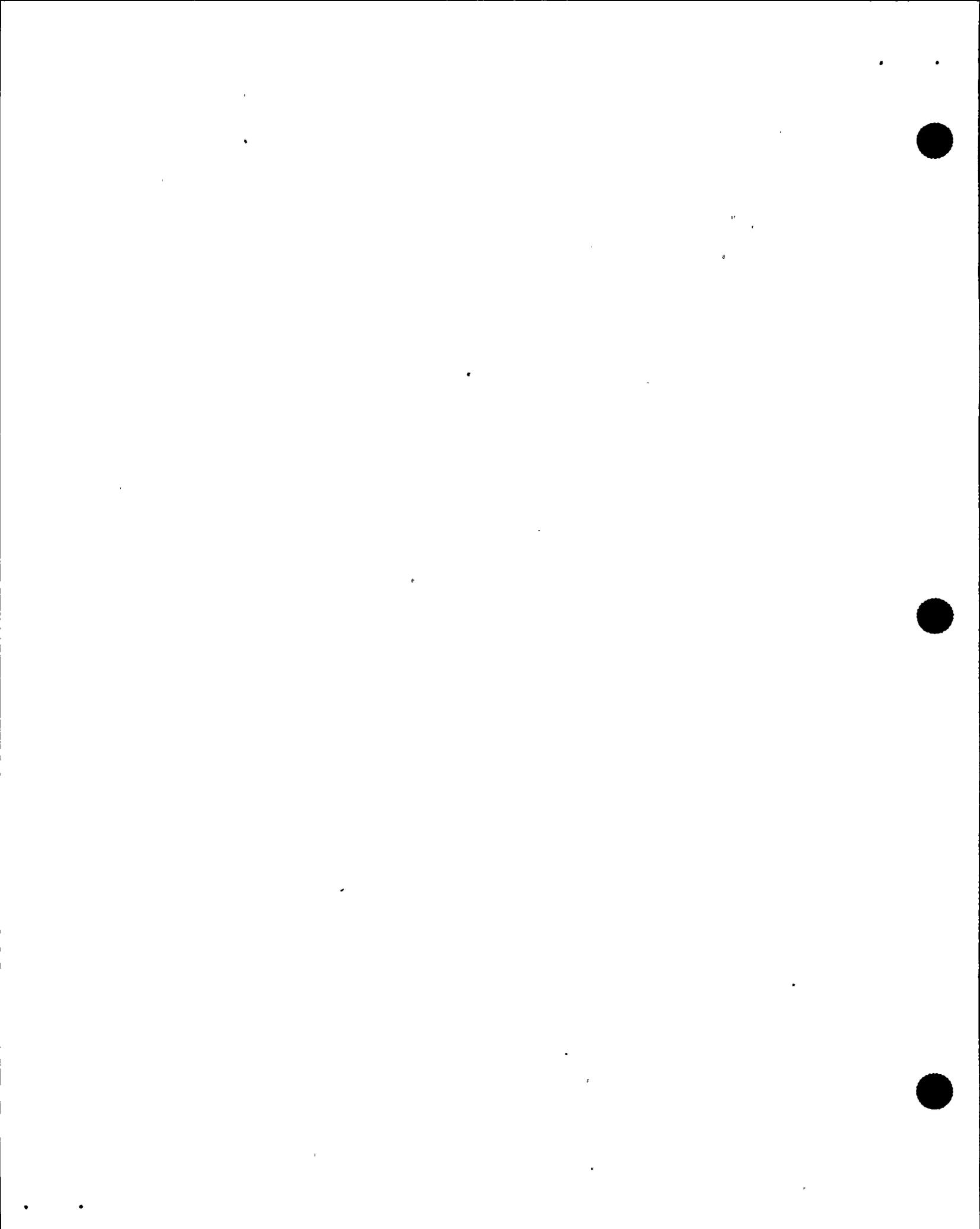


III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.



III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.



III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

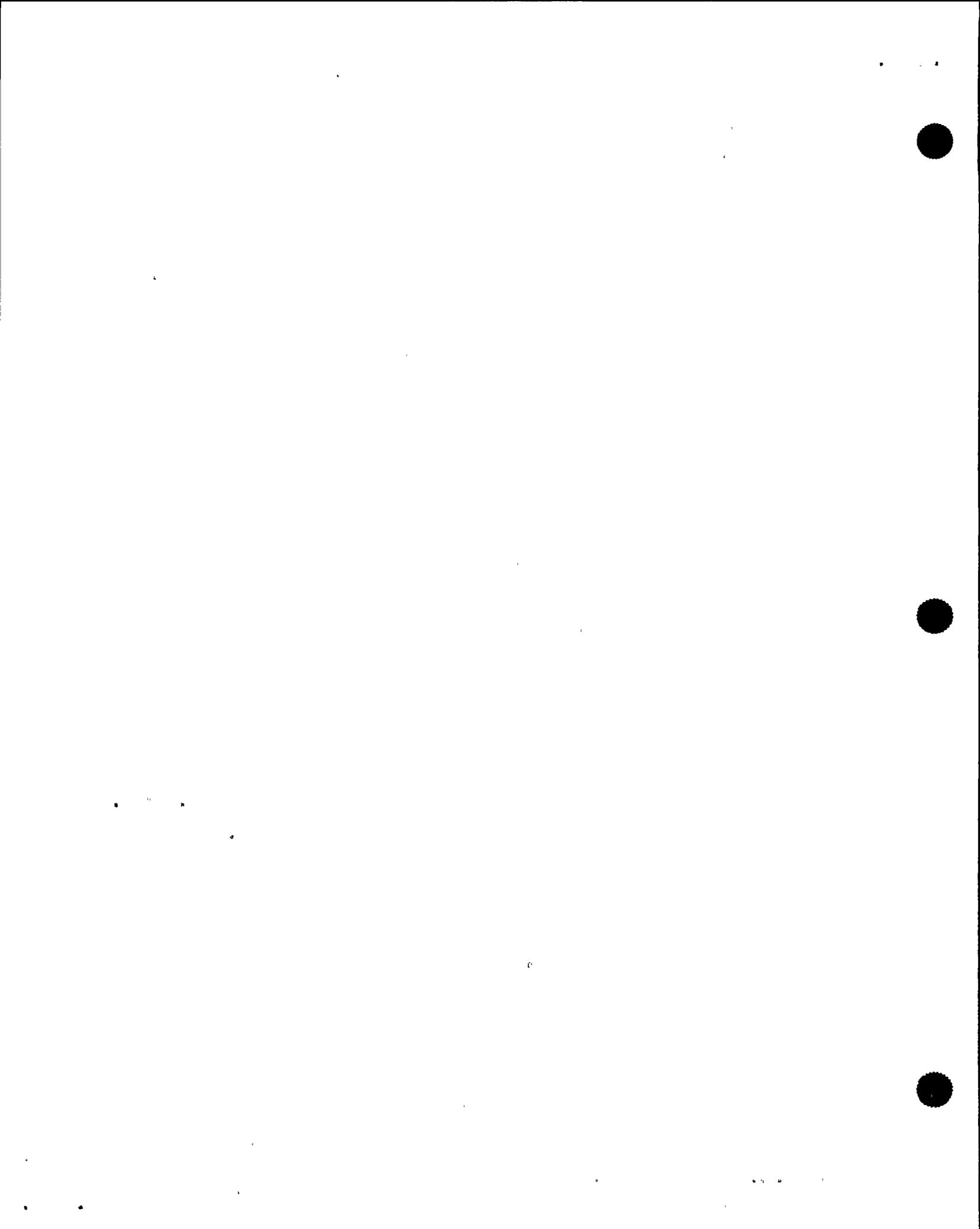
"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.



III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.



III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

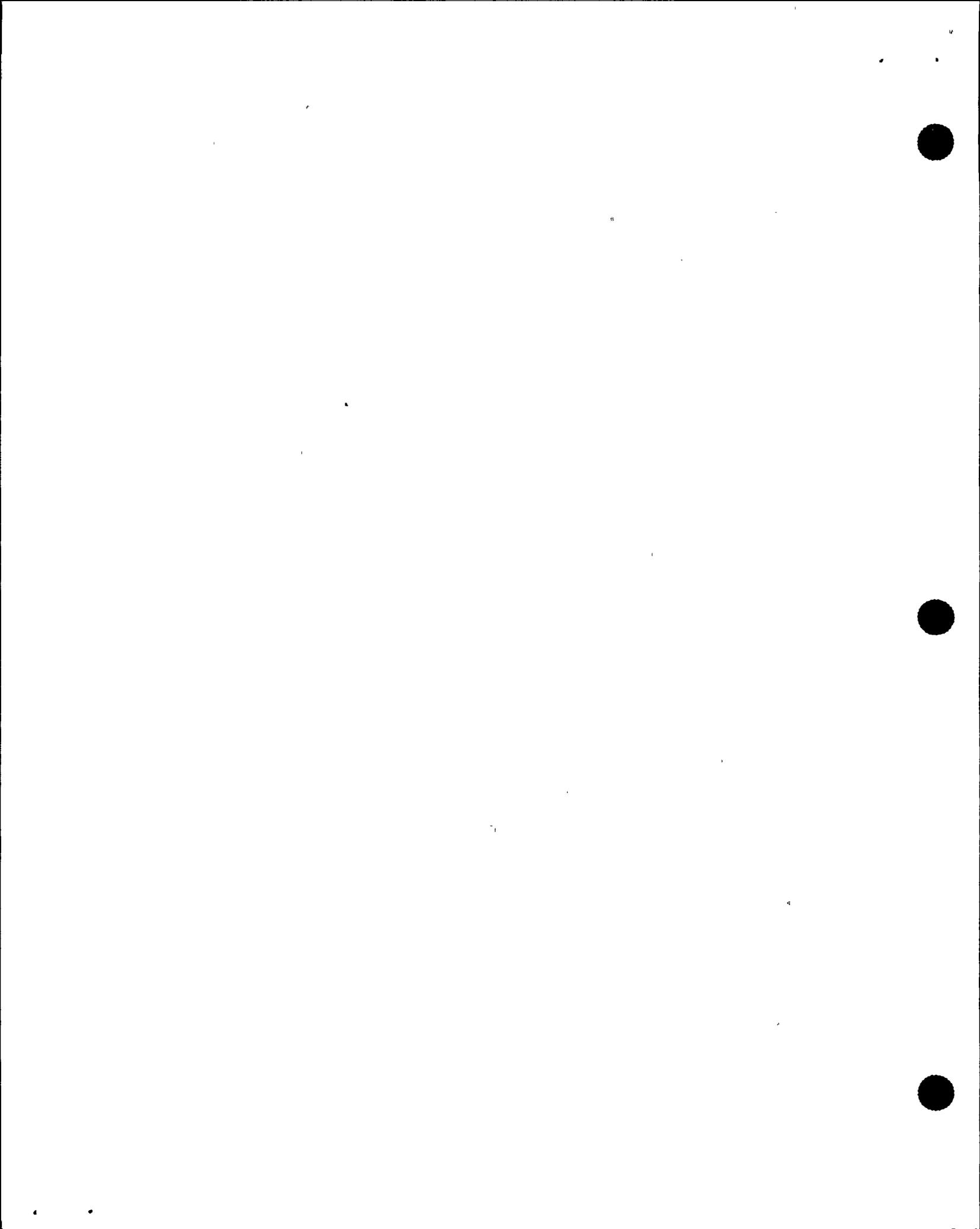
NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.



ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS



MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers (1 Page)

NUREG-1431 Specifications that are not applicable (1 Page)

Mark-up:

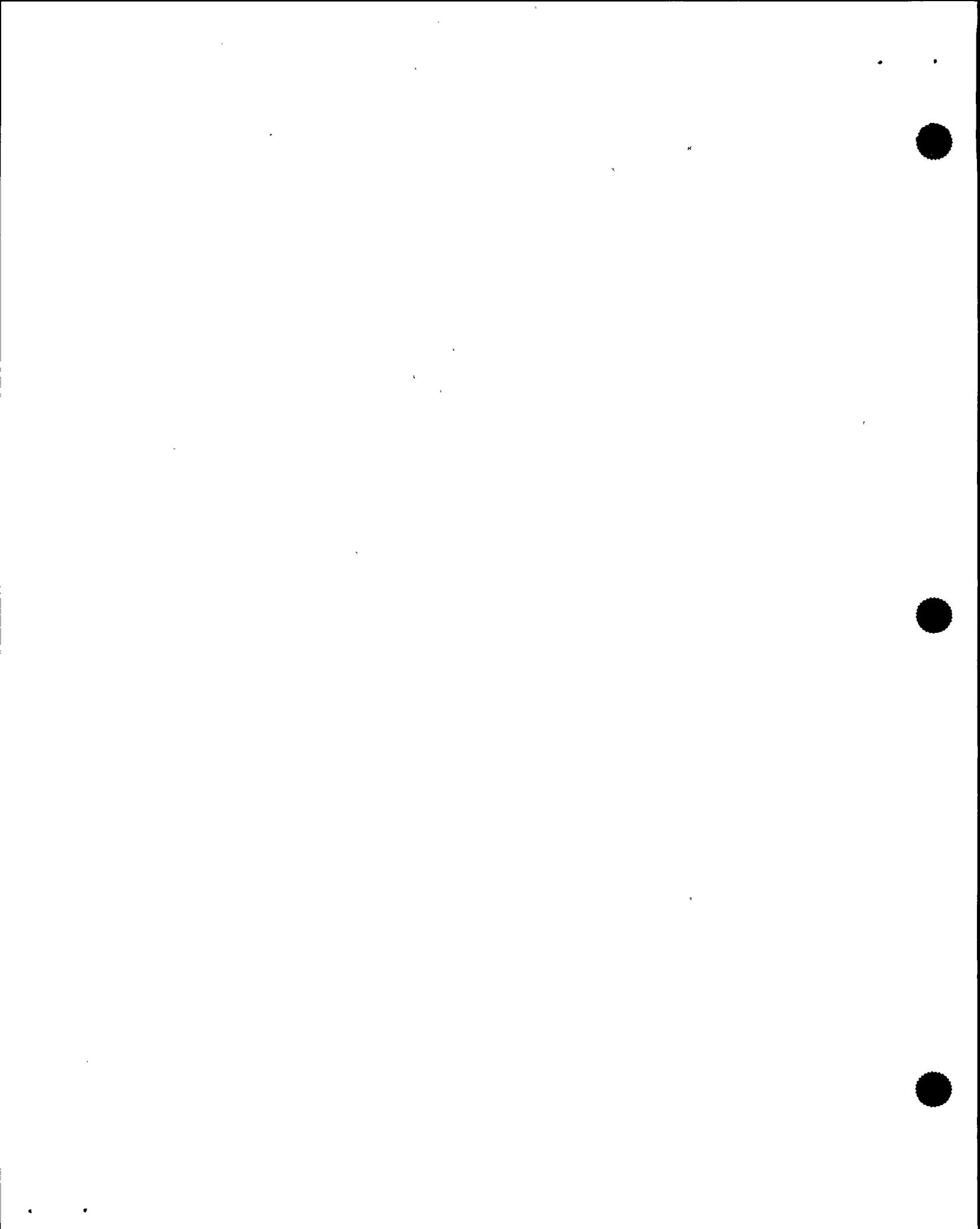
<u>SPECIFICATION</u>	<u>PAGE</u>
2.1	2.0-1
2.2	2.0-1

Methodology (2 Pages)



Industry Travelers Applicable to Section 2.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF, Rev. 1	Incorporated	2.0-1, 2.0-2	NRC approved.
TSTF-65	Not incorporated	N/A	Specifications 2.2.4 and 2.2.5 were deleted per TSTF-5, Rev. 1



NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

None



2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained \leq {2735} psig.

B

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

~~2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.~~

2.0-01

~~2.2.4 Within 24 hours, notify the [Plant Superintendent and Vice President — Nuclear Operations].~~

2.0-02

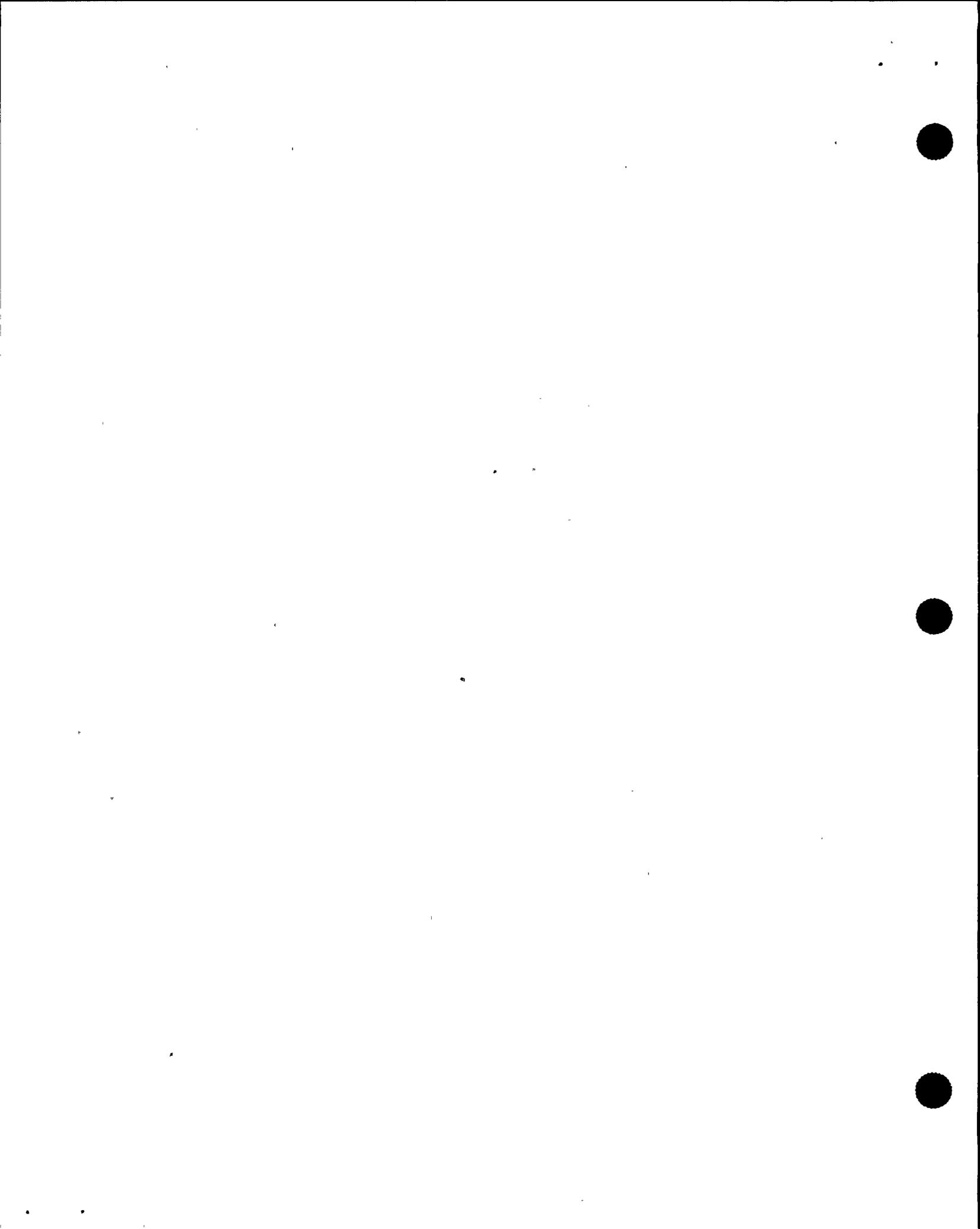
~~2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the [offsite review function], and the [Plant Superintendent, and Vice President — Nuclear Operations].~~

2.0-01

2.0-02

~~2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.~~

2.0-01



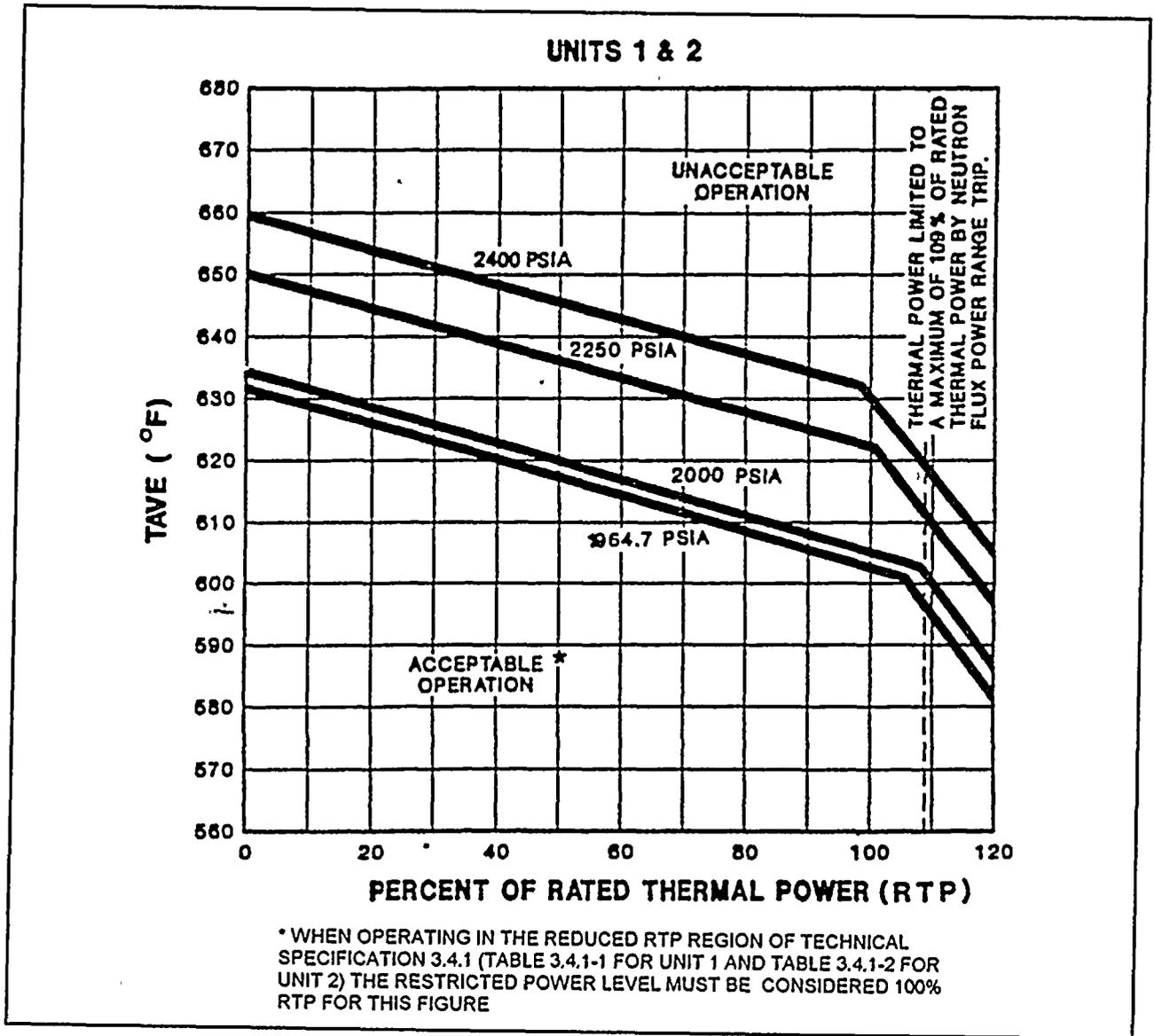
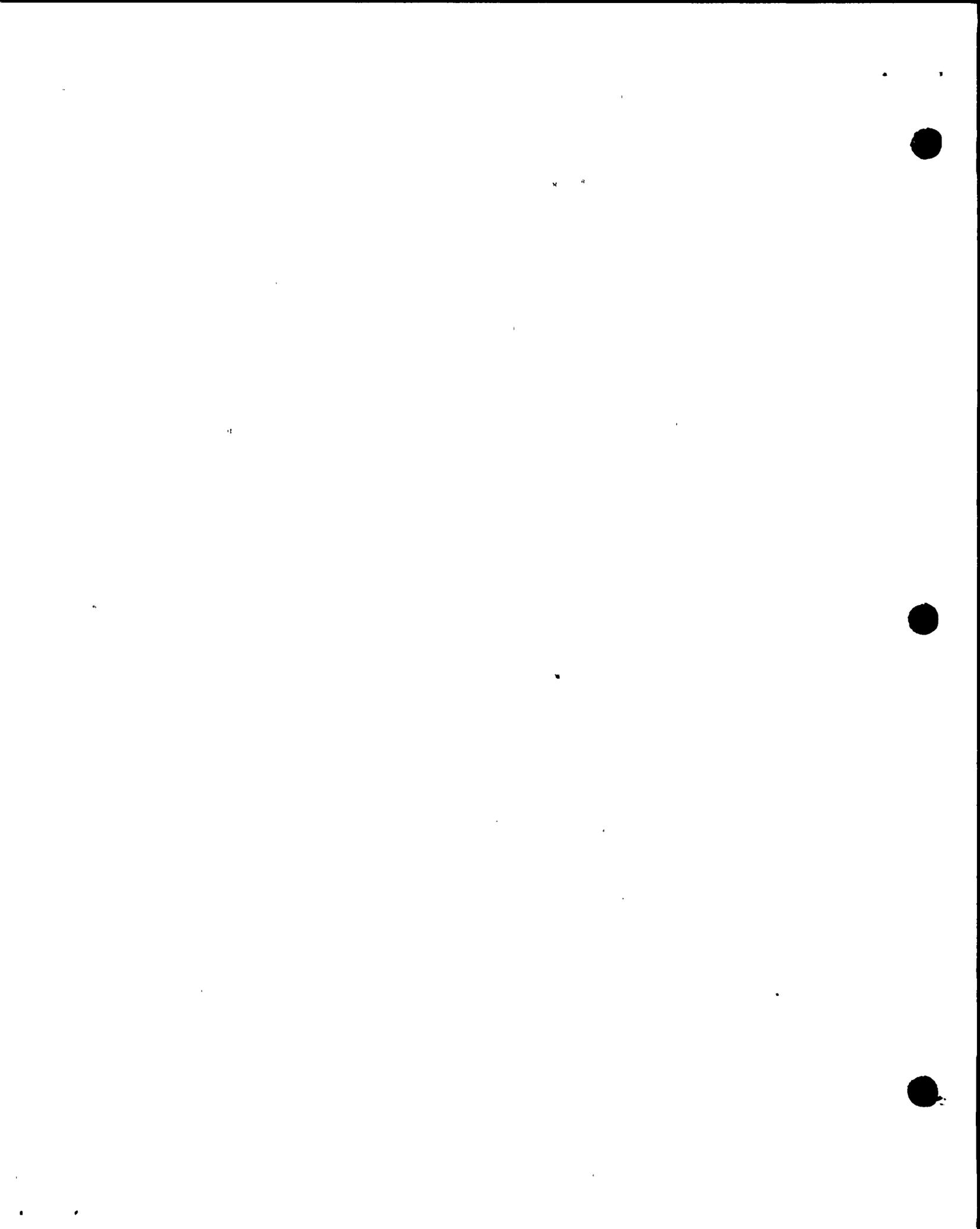


Figure 2.1.1-1



Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes- Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.



**Methodology For Mark-up of NUREG-1431 Specifications
(Continued)**

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

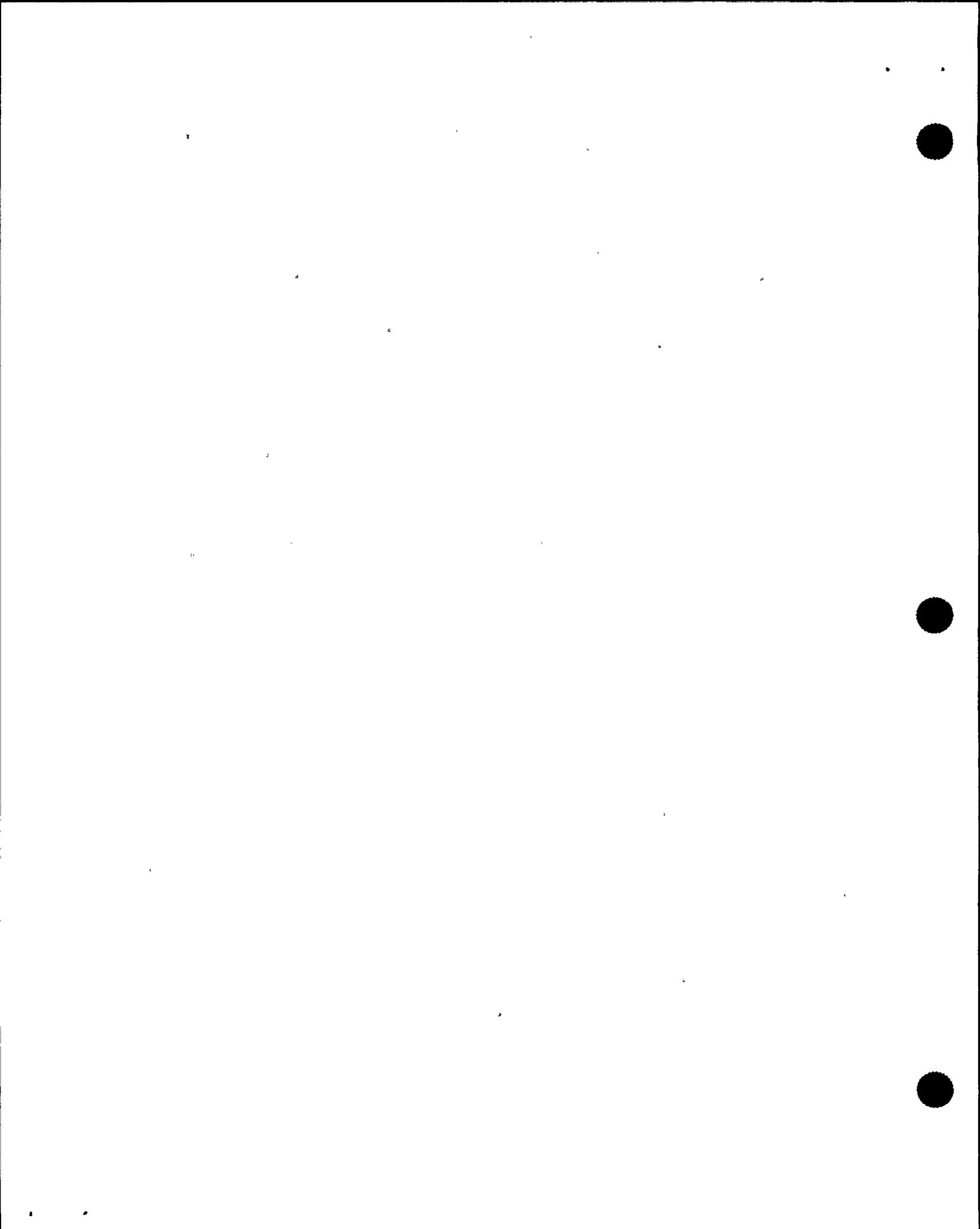
If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

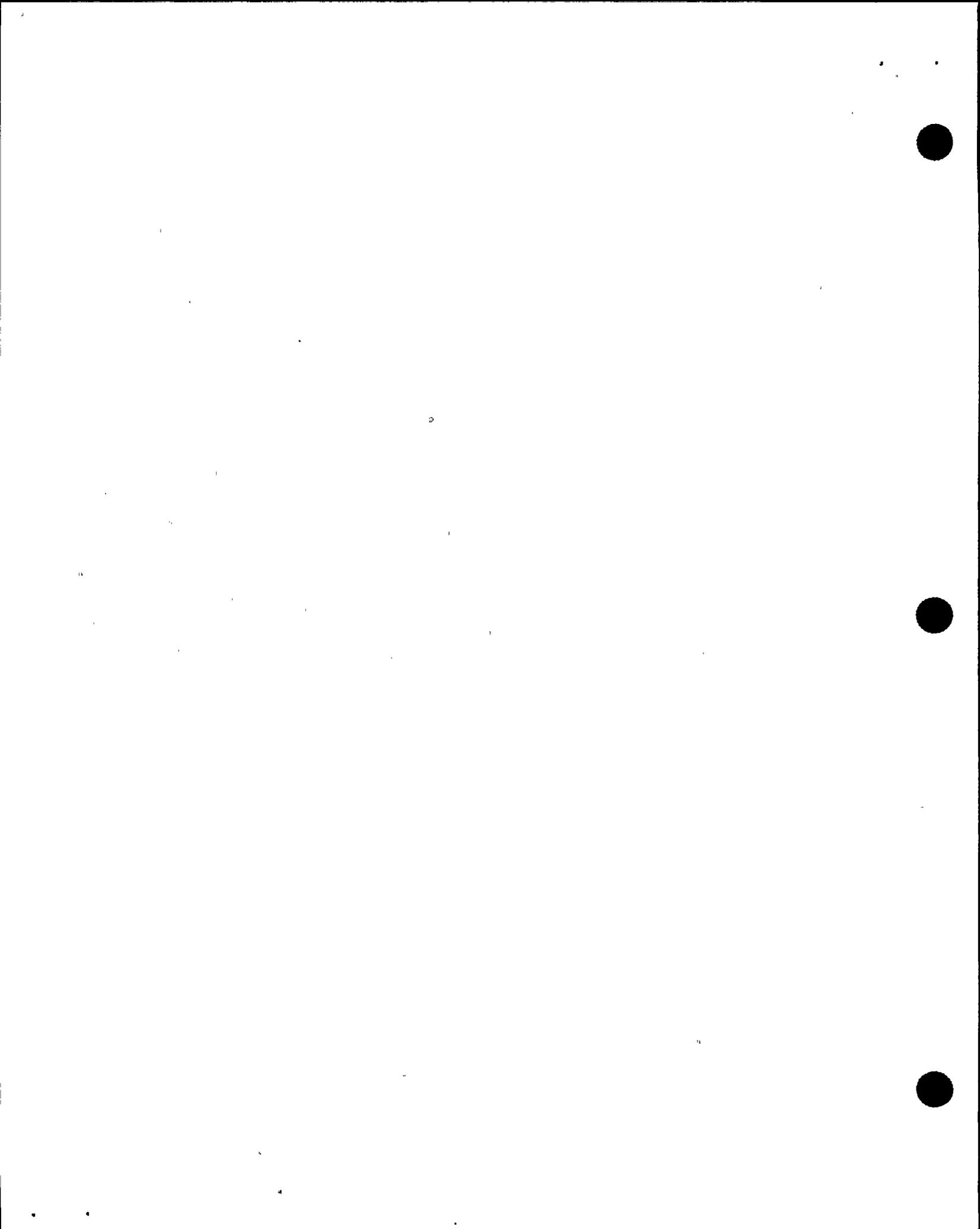
In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.



ENCLOSURE 5B

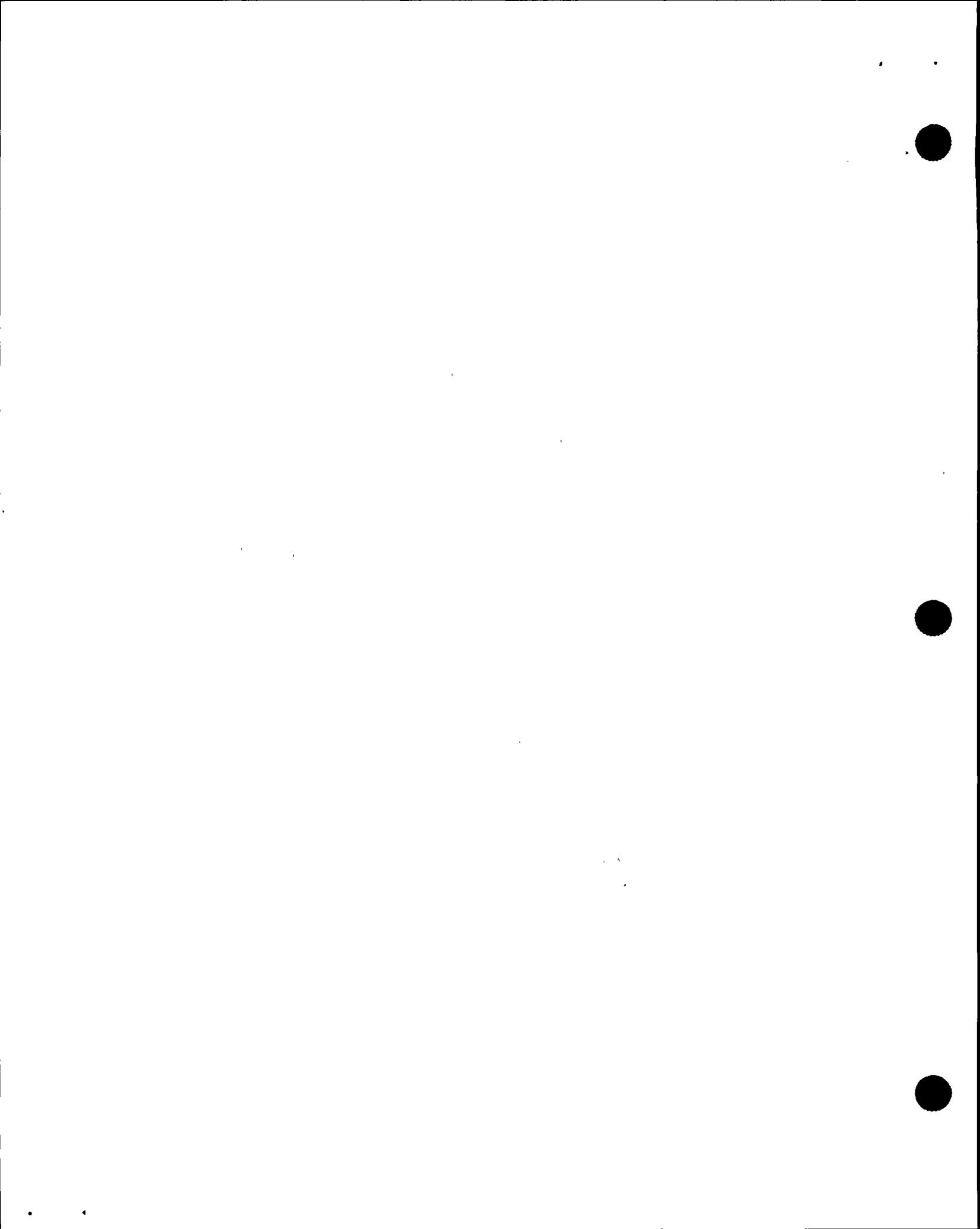
MARK-UP OF NUREG-1431 BASES



MARK-UP OF NUREG-1431 BASES CONTENTS

Mark-up:

<u>BASES</u>	<u>PAGE</u>
2.1.1	B 2.0-1
2.1.2	B 2.0-6
Methodology	(1 Page)



B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

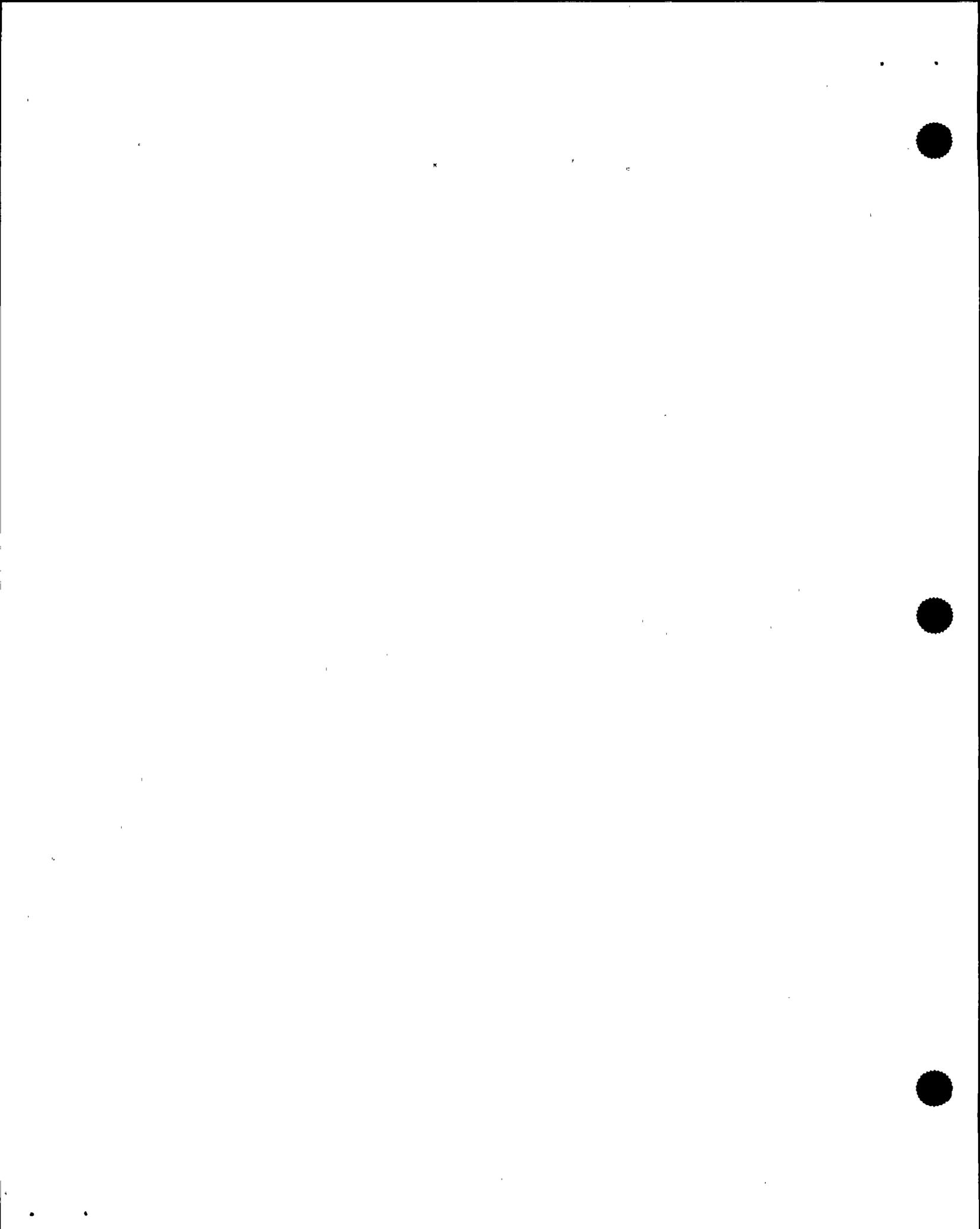
BACKGROUND GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant. The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

(Continued)



BASES (Continued)

APPLICABLE SAFETY ANALYSES The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

- a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and
- b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

~~Automatic enforcement of Protection for these reactor core SLs is provided by the steam generator safety valves and the following automatic reactor trip functions:~~ following functions:

- a. High pressurizer pressure trip;
- b. Low pressurizer pressure trip;
- c. Overtemperature ΔT trip;
- d. Overpower ΔT trip;
- e. Power Range Neutron Flux High trip; and
- f. ~~Steam generator safety valves.~~

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2 5) provide more restrictive limits to ensure that the SLs are not exceeded.

(Continued)



BASES (Continued)

SAFETY LIMITS

The curves provided in Figure B 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and average temperature for below which the minimum calculated DNBR is not less than the safety analyses limit, the design DNBR value, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The curves are based on enthalpy rise hot channel factor limits provided in the COLR. The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.

The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

~~The following SL violation responses are applicable to the reactor core SLs.~~

2.2.1

~~The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.~~

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

~~Per 10CFR50.36, if a safety limit is violated, operations must not be resumed until authorized by the Commission.~~

2.2.3

~~If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 5).~~

(Continued)



BASES (Continued)

SAFETY LIMIT
VIOLATIONS
(continued)

2.2.4

~~If SL 2.1.1 is violated, the Plant Superintendent and the Vice President Nuclear Operations shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.~~

2.2.5

~~If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 6). A copy of the report shall also be provided to the Plant Superintendent and the Vice President Nuclear Operations.~~

2.2.6

~~If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~

-
- REFERENCES
1. 10 CFR 50, Appendix A, GDC 10.
 2. FSAR, Section [7.2] ~~Chapter 7~~.
 3. WCAP-8746-A, March 1977.
 4. ~~WCAP 9273 NP A, July 1985.~~
 5. ~~10 CFR 50.72. FSAR, Chapter 15.~~
 6. ~~10 CFR 50.73.~~
-



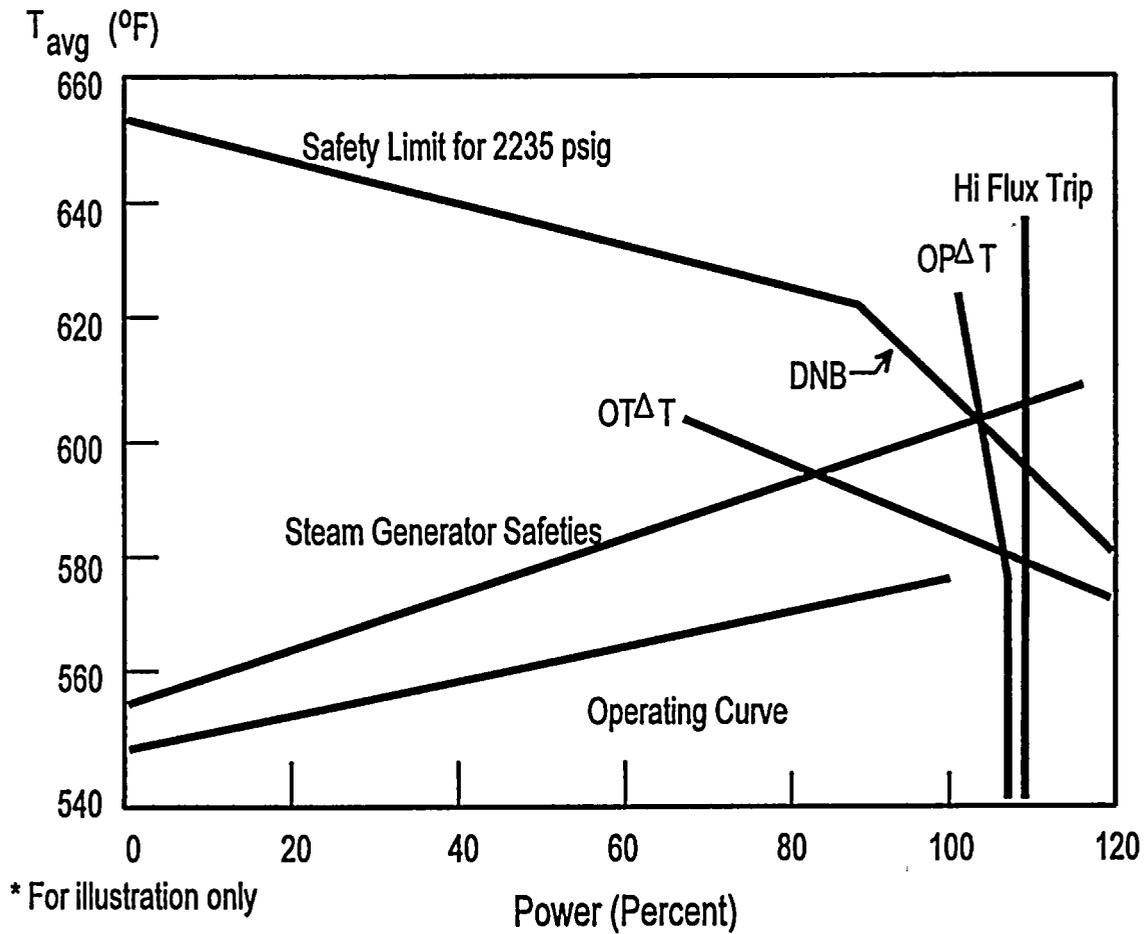
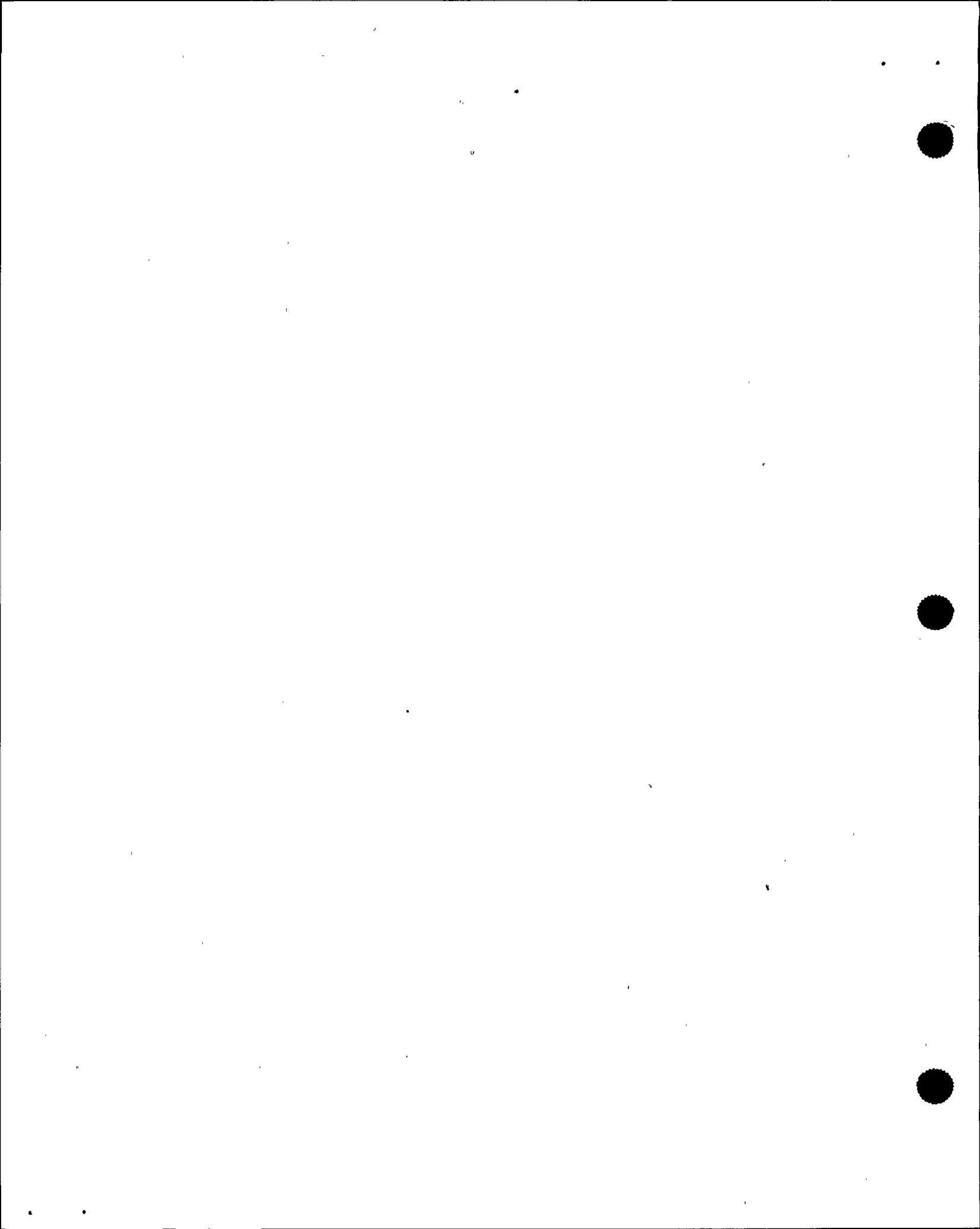


Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection



B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

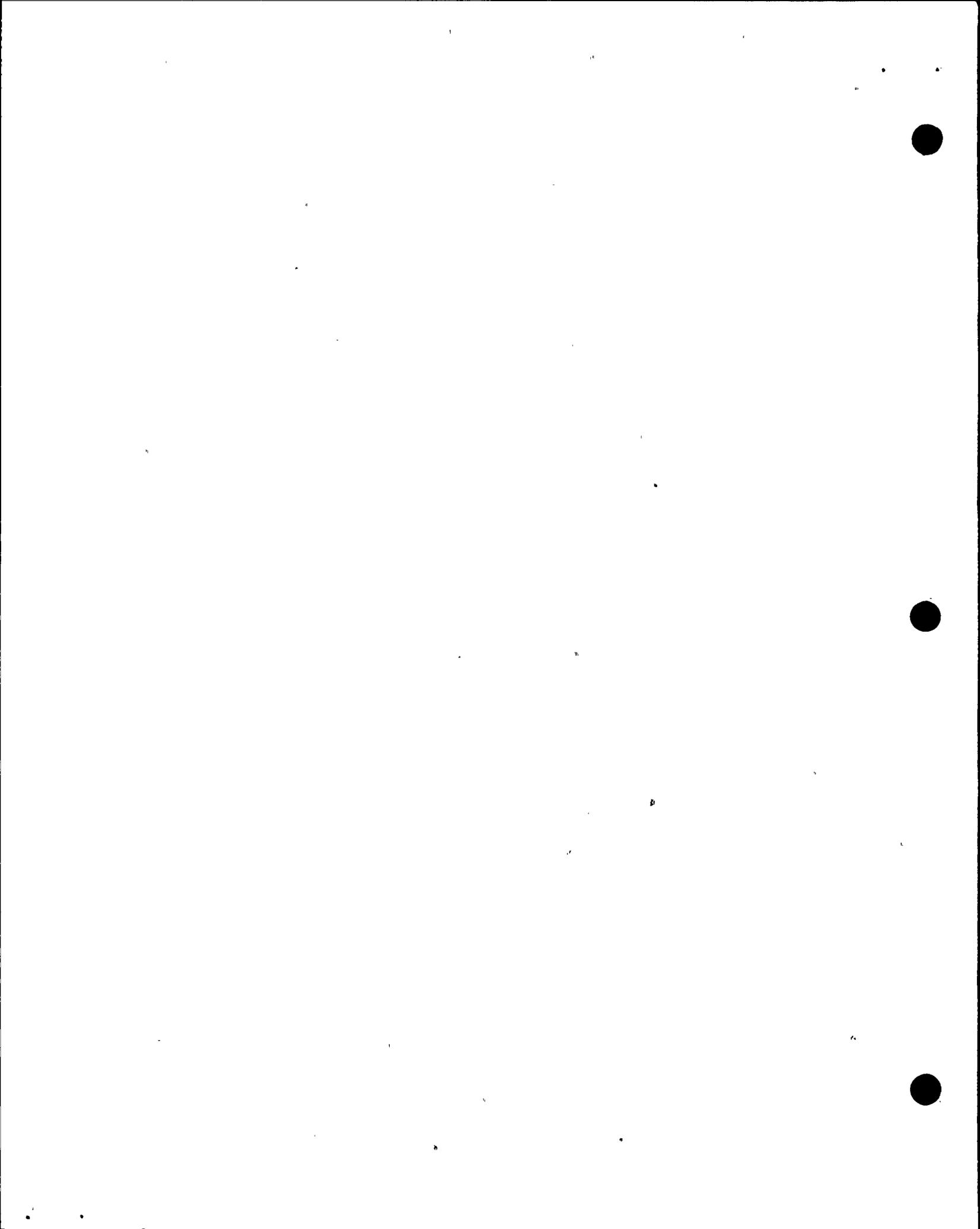
The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2485 psig ~~2500 psia~~. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components ~~are~~ were hydrostatically tested at ~~125% 150% (3750)~~ (Ref. 9) of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE
SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.



BASES (Continued)

APPLICABLE
SAFETY ANALYSES
(continued)

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load turbine trip without a direct reactor trip. During the transient, no control actions are assumed, except that the Safety valves on the secondary plant side are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal lost at the time of turbine trip main feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 5) allowable values, together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

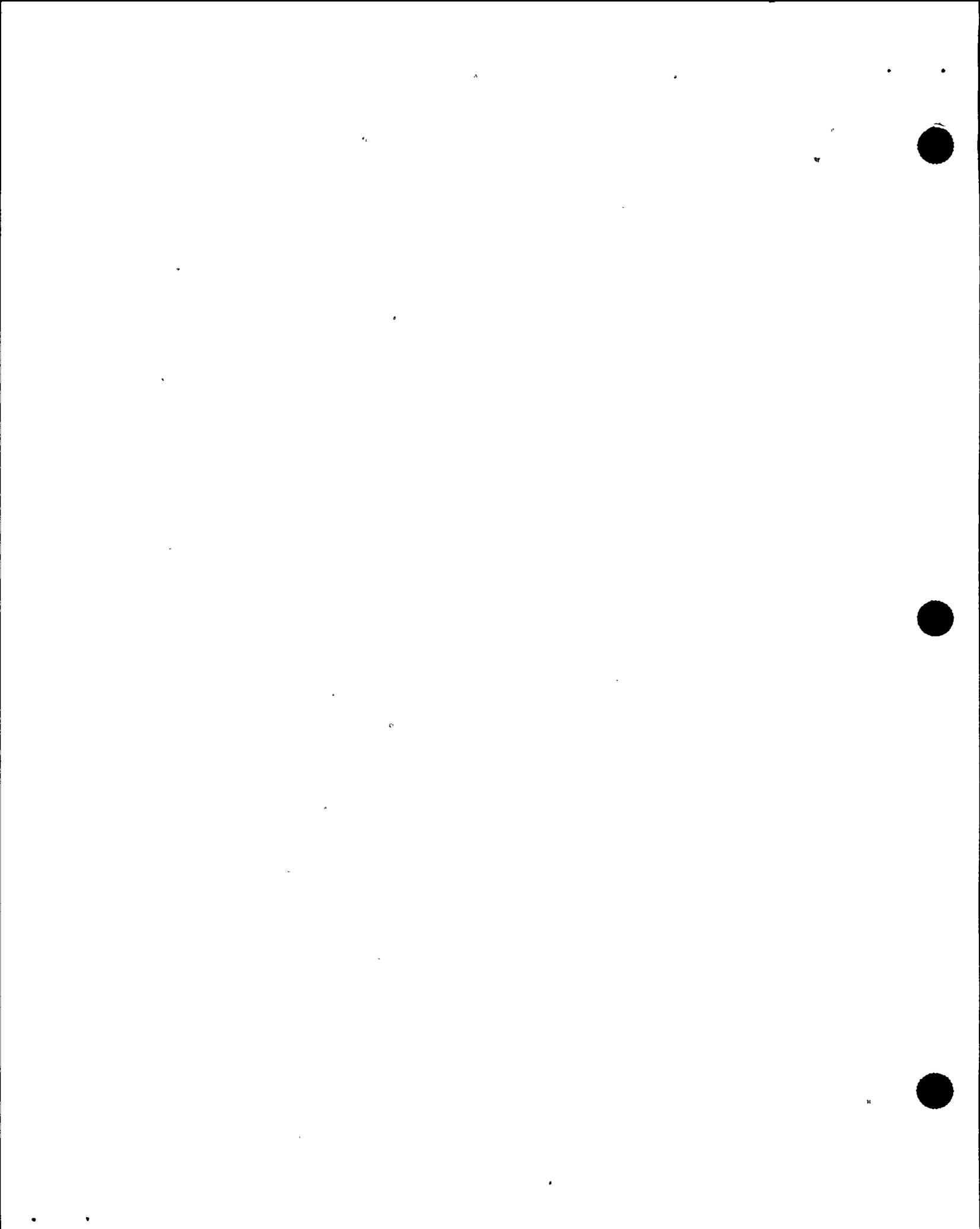
- a. Pressurizer power operated relief valves (PORVs);
- b. Steam line relief valve Generator Atmospheric Dump valves;
- c. Steam Dump System;
- d. Reactor Control System;
- e. Pressurizer Level Control System; or
- f. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under [USAS, Section B31.1 (Ref. 6)] is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, The SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, or the reactor vessel is sufficiently vented making it unlikely that the RCS can be pressurized.



BASES (Continued)

SAFETY LIMIT
VIOLATIONS

~~The following SL violations are applicable to the RCS pressure SL.~~

2.2.2.1

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

2.2.2.2

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.2.3

~~If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).~~

2.2.4

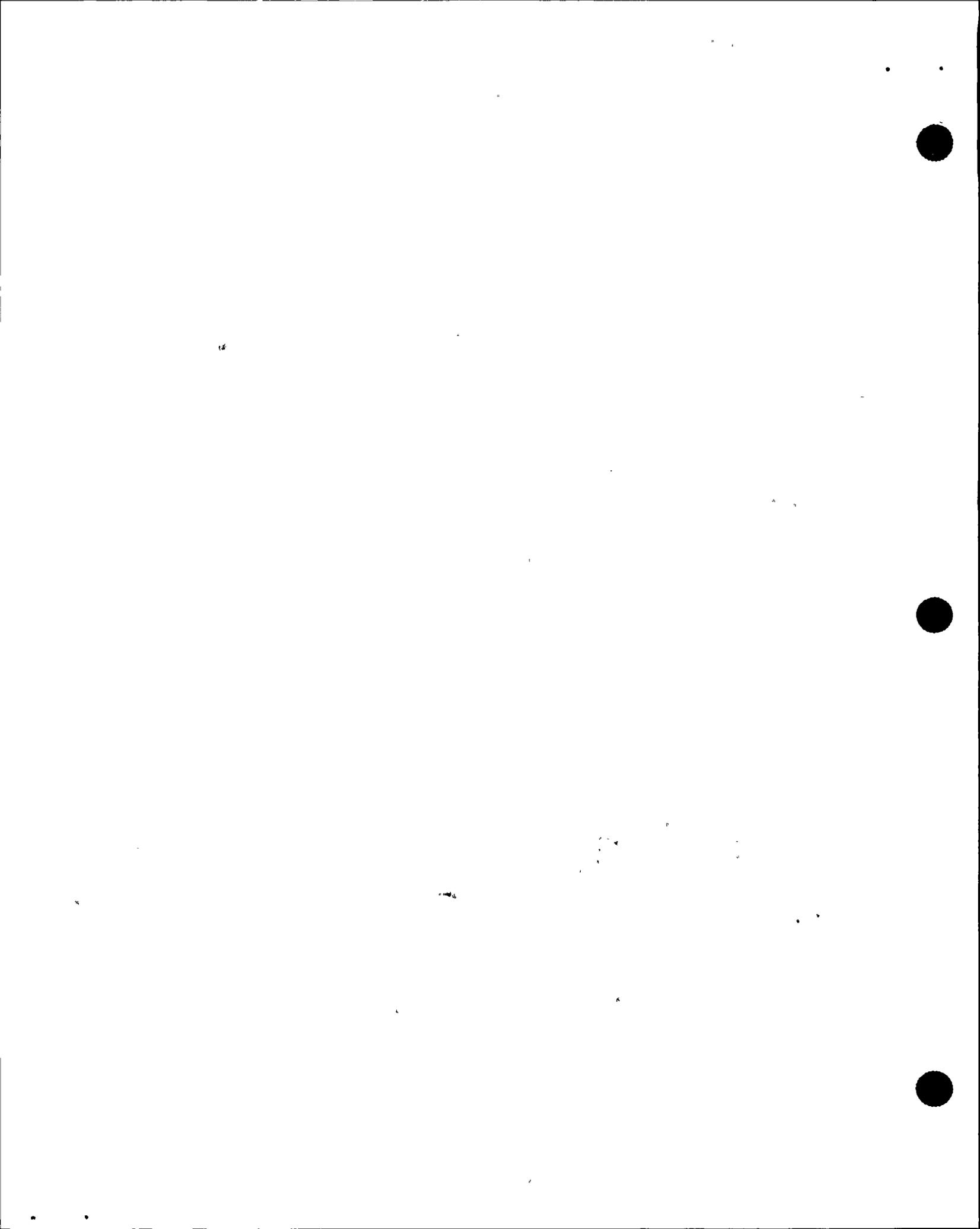
~~If the RCS pressure SL is violated, the Plant Superintendent and the Vice President Nuclear Operations shall be notified within 24 hours. The 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.~~

2.2.5

~~If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the Plant Superintendent and the Vice President Nuclear Operations.~~

2.2.6

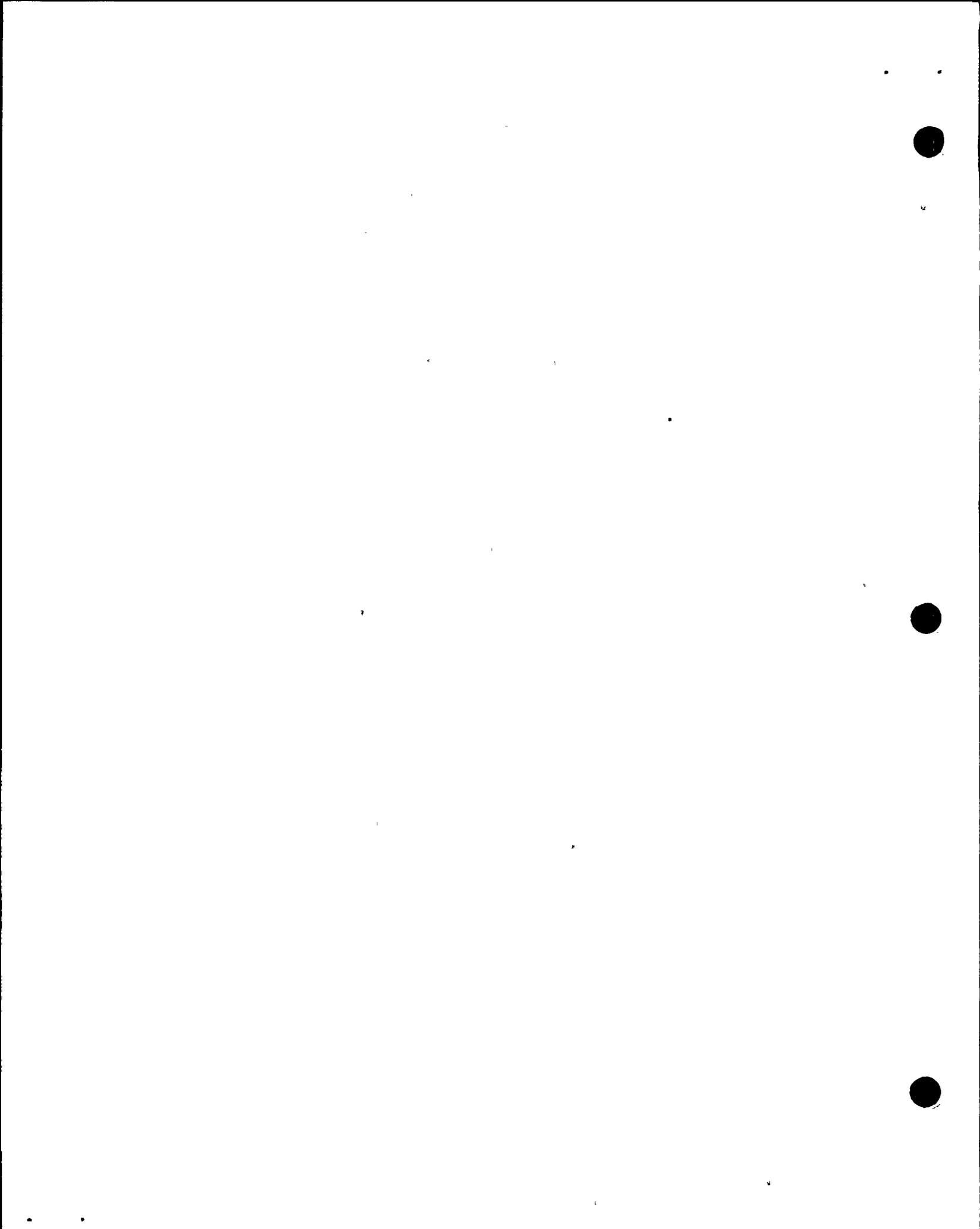
~~If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~



BASES (Continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
 2. ~~ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.~~
 3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
 4. 10 CFR 100.
 5. ~~FSAR, Section [7.2].~~
 6. ~~USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.~~
 7. ~~10 CFR 50.72.~~
 8. ~~10 CFR 50.73.~~
 9. Westinghouse report SD-117, "Structural Analysis of Reactor Coolant Loop/Support System for Diablo Canyon Nuclear Generating Station Unit No. 1, February, 1975."
-
-



Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific. To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

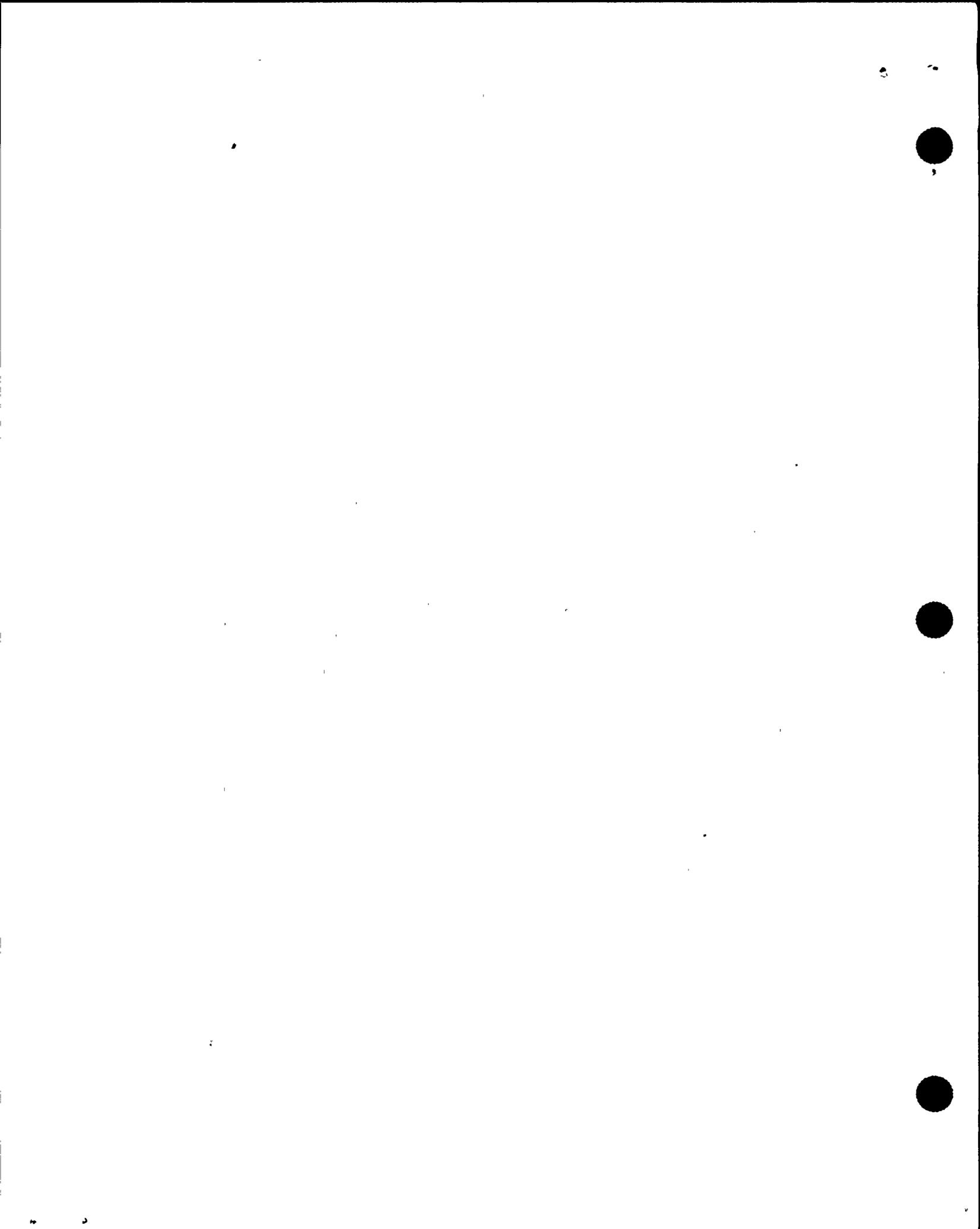


ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(1 Page)



JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 2.0

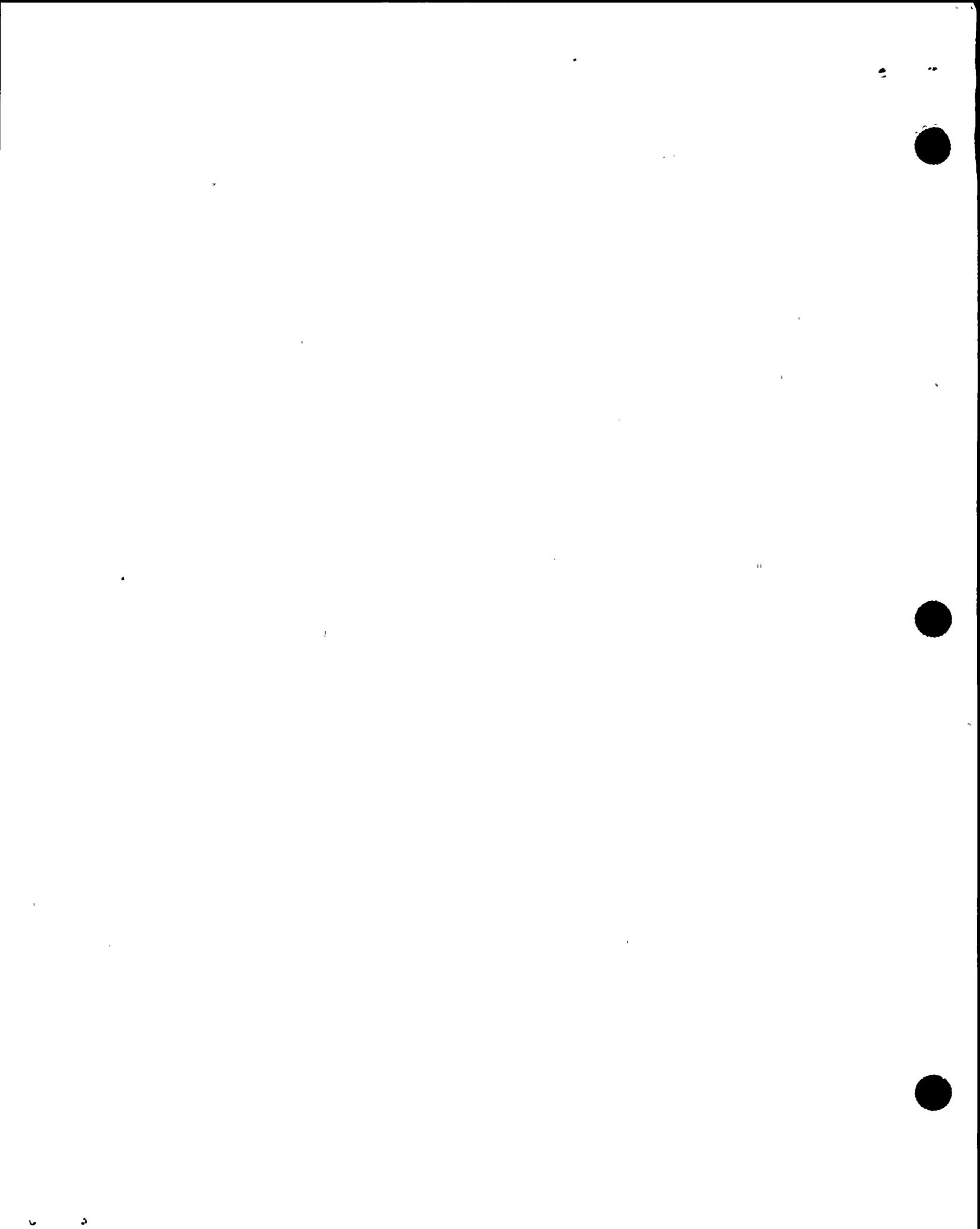
This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE

NUMBER

JUSTIFICATION

- | | |
|--------|--|
| 2.0-01 | SL violation requirements that duplicate regulations would be eliminated. The applicable regulations are 50.36, 50.72, and 50.73 for reporting an SL violation and 50.36 for obtaining NRC authorization to operate the reactor following an SL violation. These changes are in accordance with TSTF-5, Rev. 1 which has been approved by the NRC. |
| 2.0-02 | The requirements regarding notification of licensee management following SL violations and obtaining reviews of LERs by management and the offsite review committee would be deleted. These requirements would be contained in licensee controlled documents. These changes are also in accordance with TSTF-5, Rev. 1. |

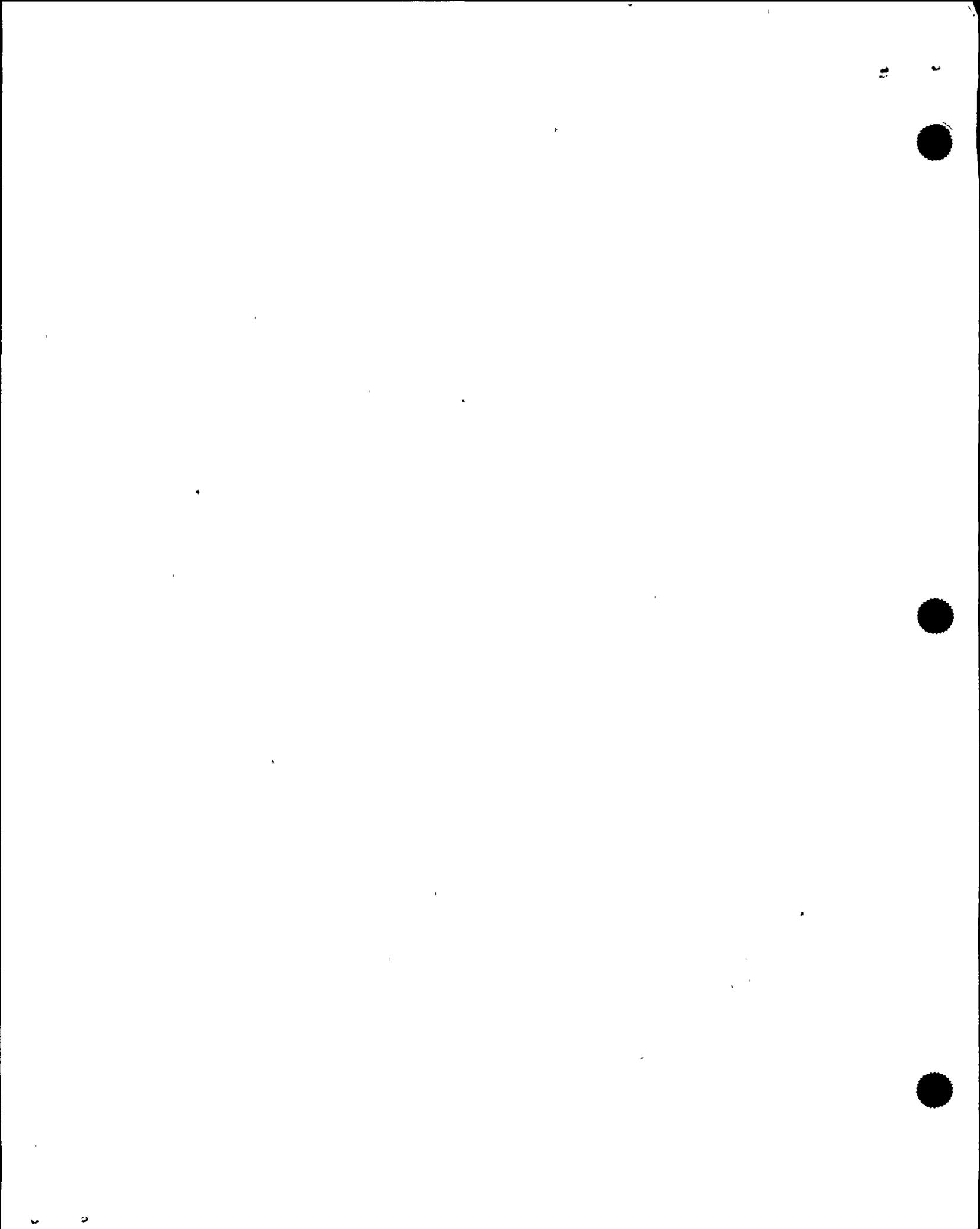


ENCLOSURE 6B

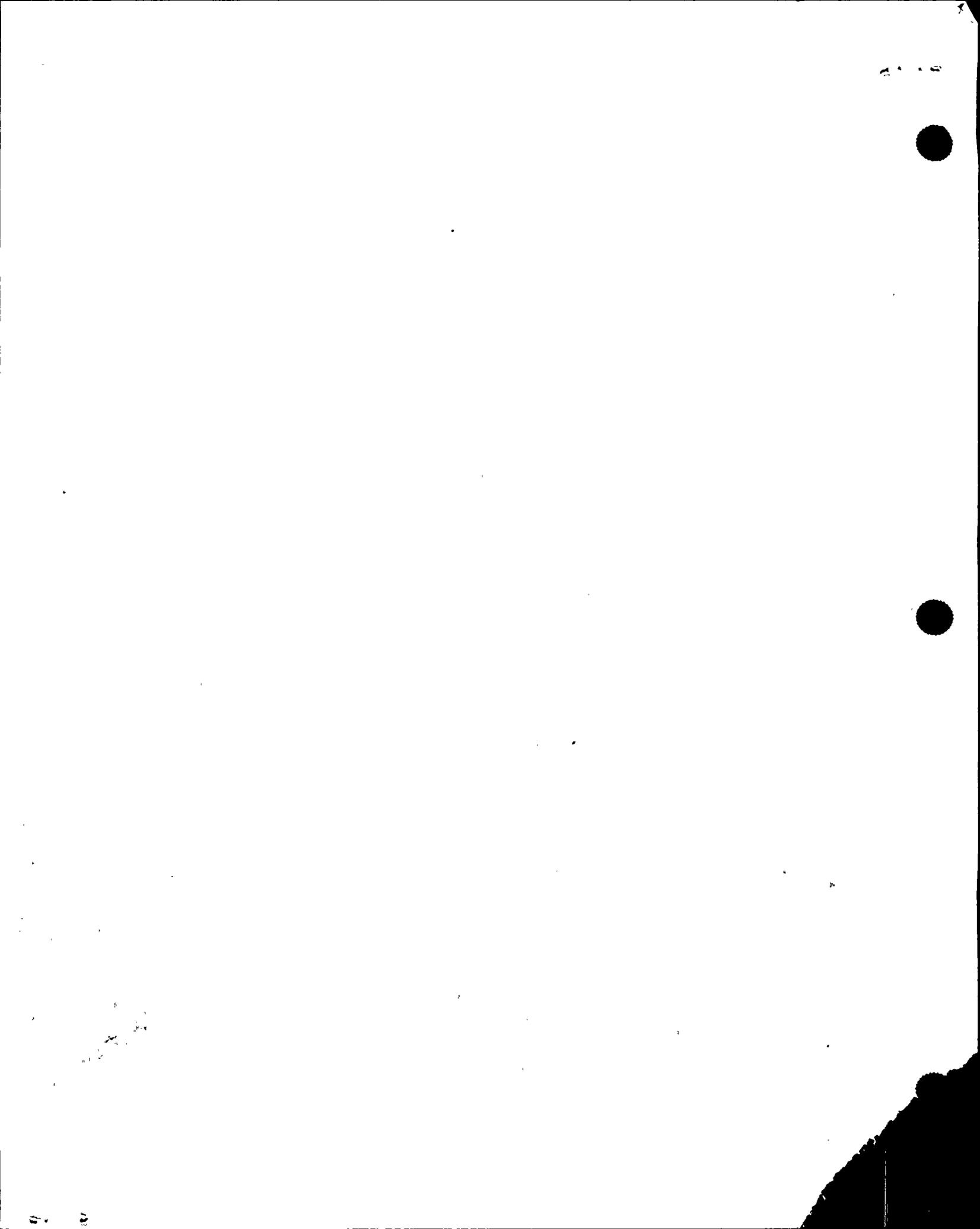
CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(1 Page)



TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
2.0-01	SL violation requirements that duplicate regulations would be eliminated.	Yes	Yes	Yes	Yes
2.0-02	The requirements regarding notification of licensee management following SL violations and obtaining reviews of LERs by management and the offsite review committee would be relocated to licensee controlled documents	Yes {relocated to FSAR Chapter 16}	Yes	Yes	Yes



JLS Conversion to Improved Technical Specifications Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 3/4.0 - Limiting Conditions for Operation and
Surveillance Requirements

ITS 3.0 - Limiting Conditions for Operation
Applicability/Surveillance Requirement
Applicability



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.0

CONTENTS

- ENCLOSURE 1 - CROSS-REFERENCE TABLES
- ENCLOSURE 2 - MARK-UP OF CURRENT TS
- ENCLOSURE 3A - DESCRIPTION OF CHANGES TO CURRENT TS
- ENCLOSURE 3B - CONVERSION COMPARISON TABLE - CURRENT TS
- ENCLOSURE 4 - NO SIGNIFICANT HAZARDS CONSIDERATIONS
- ENCLOSURE 5A - MARK-UP OF NUREG-1431 SPECIFICATIONS
- ENCLOSURE 5B - MARK-UP OF NUREG-1431 BASES
- ENCLOSURE 6A - DIFFERENCES FROM NUREG-1431
- ENCLOSURE 6B - CONVERSION COMPARISON TABLE - NUREG 1431

ENCLOSURE 1

CROSS-REFERENCE TABLES

8

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(1 Page)
CONVERSION TABLE SORTED BY IMPROVED TS	(1 Page)
METHODOLOGY	(3 Pages)

CROSS-REFERENCE TABLE FOR 3/4.0
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.0.1	LCO		01-01-A	3.0.1	LCO		3.0-01
3.0.2	LCO		01-01-A	3.0.2	LCO		
3.0.3	LCO			3.0.3	LCO		
3.0.4	LCO		01-02-LS1	3.0.4	LCO		3.0-03
3.0.5	LCO		01-03-A			Not Used	
3.0.5	LCO		01-04-LS2	3.0.5	LCO		
		New	01-05-M	3.0.6	LCO		3.0-04
		New	01-06-A	3.0.7	LCO		3.0-02
4.0.1	SR		01-20-A	3.0.1	SR		
4.0.2	SR		01-07-LS3 01-19-A	3.0.2	SR		
4.0.3	SR		01-08-LS4 01-20-A	3.0.3	SR		
4.0.4	SR		01-09-LS1 01-19-A	3.0.4	SR		
4.0.5	SR		01-10-A	5.5.8			5.5-05
4.0.5	SR	a	01-10-A 01-11-LG	5.5.8			
4.0.5	SR	b	01-12-A	5.5.8			
4.0.5	SR	c	01-10-A	5.5.8			
4.0.5	SR	d	01-17-A	5.5.8			
4.0.5	SR	e				Not Used	
4.0.5	SR	New	01-13-A	5.5.8			
4.0.6	SR		01-03-A			Not Used	

CROSS-REFERENCE TABLE FOR 3/4.0
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.0.1	LCO		01-01-A	3.0.1	LCO		3.0-01
3.0.2	LCO		01-01-A	3.0.2	LCO		
3.0.3	LCO			3.0.3	LCO		
3.0.4	LCO		01-02-LS1	3.0.4	LCO		3.0-03
		New	01-04-LS2	3.0.5	LCO		
		New	01-05-M	3.0.6	LCO		3.0-04
		New	01-06-A	3.0.7	LCO		3.0-02
4.0.1	SR		01-20-A	3.0.1	SR		
4.0.2	SR		01-07-LS3 01-19-A	3.0.2	SR		
4.0.3	SR		01-08-LS4 01-20-A	3.0.3	SR		
4.0.4	SR		01-09-LS1 01-19-A	3.0.4	SR		

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

**Methodology for Cross-Reference Tables
(Continued)**

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).
Note:	The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
3/4.0 Applicability	
Limiting Condition For Operation	3/4 0-1
Surveillance Requirements	3/4 0-2
Methodology	(2 Pages)

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, except as provided in LCO 3.0.2 and LCO 3.0.7. 01-01-A

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals, except as provided in LCO 3.0.5 and LCO 3.0.6. If the Limiting Condition for Operation is restored or is no longer applicable prior to expiration of the specified time intervals, completion of the ACTION requirements is not required unless otherwise stated. 01-01-A

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

- a. At least HOT STANDBY within the next 6 hours,
- b. At least HOT SHUTDOWN within the following 6 hours, and
- c. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions of these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION statements or that are part of a shutdown of the unit. Exceptions to these requirements are stated in the individual specifications. 01-02-ESI

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODE 1, 2, 3, and 4. 01-02-ESI

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

~~3.0.5 Limiting Conditions for Operation including the associated ACTION requirements shall apply to each unit individually unless otherwise indicated as follows:~~

- ~~a- Whenever the Limiting Conditions for Operation refers to systems or components which are shared by both units, the ACTION requirements will apply to both units simultaneously. This will be indicated in the ACTION section;~~ 01-03-A
- ~~b- Whenever the Limiting Conditions for Operation applies to only one unit, this will be identified in the APPLICABILITY section of the specification; and~~
- ~~c- Whenever certain portions of a specification contain operating parameters, Setpoints, etc., which are different for each unit, this will be identified in parentheses, footnotes or body of the requirement.~~

~~(new) Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.~~ 01-04-ES2

~~(new) When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.15, Safety Function Determination Program (SFDP). If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.~~ 01-05-M

~~When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.~~

~~(new) ITS Test Exception LCO 3.1.8 allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.~~ 01-06-A

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

For Frequencies specified as "once," the above interval extension does not apply.

01-07-LS3

If a Completion Time requires periodic performance on a "once per" basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this requirement are stated in the individual specifications.

01-19-A

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval defined by Specification 4.0.2 shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed from the time of discovery, for up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. Exceptions to these requirements are stated in the individual specifications. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

01-08-LS4

Surveillances Requirements do not have to be performed on inoperable equipment or variables outside specified limits.

01-20-A

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements or that are part of a shutdown of the unit.

01-09-LS1

SR 4.0.4 is only applicable for entry into a MODE or other specified Condition in the APPLICABILITY in MODES 1, 2, 3, and 4.

01-09-LS1

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components shall be applicable as follows:

- a. ~~Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps, valves.~~

01-11-EG

01-10-A

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

~~and snubbers shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(I);~~

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice ~~inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:~~

01-11-EG

ASME BOILER AND PRESSURE VESSEL
CODE AND APPLICABLE ADDENDA
TERMINOLOGY FOR INSERVICE
INSPECTION AND TESTING ACTIVITIES

REQUIRED FREQUENCIES FOR
PERFORMING INSERVICE
INSPECTION AND TESTING
ACTIVITIES

01-11-EG

Weekly

At least once per 7 days

Monthly

At least once per 31 days

Quarterly or every 3 months

At least once per 92 days

Semiannually or every 6 months

At least once per 184 days

Every 9 months

At least once per 276 days

Yearly or annually

At least once per 366 days

~~Biennially or every 2 years~~

~~At least once per 731 days~~

01-12-A

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

- c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities; 01-10-A
- d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements; and 01-17-A
- e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

~~The provisions of 4.0.3 are applicable to IST activities~~

01-13-A

~~4.0.6 Surveillance Requirements shall apply to each unit individually unless otherwise indicated as stated in Specification 3.0.5 for individual specifications or whenever certain portions of a specification contain surveillance parameters different for each unit, which will be identified in parentheses, footnotes or body of the requirement.~~

01-03-A

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. Deletions - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. Additions - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. Modifications - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. Administrative - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

**Methodology For Mark-Up of Current TS
(Continued)**

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Description of Changes (5 Pages)

DESCRIPTION OF CHANGES TO TS SECTION 3/4.0

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	A	Adds exceptions to CTS Limiting Condition for Operation (LCO) 3.0.1 and LCO 3.0.2 as provided in improved Technical Specifications (ITS) LCOs 3.0.5, 3.0.6, and 3.0.7. The effects of this change will be discussed later under the discussion of the addition of these LCOs. The exception to CTS LCO 3.0.1 regarding ITS LCO 3.0.7 is in accordance with industry Traveler TSTF-06, Rev. 1 included clarification that completion of an ACTION is not required when the LCO is no longer applicable [].
01-02	LS1	CTS LCO 3.0.4 was previously applicable for entry into all MODES. The specification has been revised to not prevent unit shutdowns and will apply only to entry into MODES 1, 2, 3, and 4 from lower MODES. This change is less restrictive in that it will allow MODE changes from either direction into MODES 5 and 6 while operating within an ACTION with a limited time window. This change is also less restrictive in that MODE changes which are part of a unit shutdown are not prevented. As required in the "Reviewer's Note" of NUREG-1431, LCO 3.0.4, a matrix is provided (see LS1) which documents the plant-specific review of all specifications for determination of where specific restrictions on MODE changes or Required Actions should be included in individual LCOs. The change is acceptable for those specific ACTIONS for which specific restrictions were not deemed warranted.
01-03	A	CTS LCO 3.0.5 [and associated Surveillance Requirement (SR) 4.0.6 are] being deleted. The conventions established by this LCO [and SR] are not needed and are no longer used. Through experience gained at two unit operation and the improved guidance of Specification 1.3, Completion Times, and the Bases discussion of Section 3.0, further clarification in the form of an additional LCO [and SR] is not warranted. This deletion is consistent with NUREG-1431, which does not include the current LCO 3.0.5 or SR 4.0.6.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.0
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-04	LS2	ITS LCO 3.0.5 would be added to the CTS in conformance with NUREG-1431. LCO 3.0.5 provides for returning to service under administrative control, equipment previously declared inoperable or removed from service, for the purpose of demonstrating its OPERABILITY or the OPERABILITY of other equipment. This is acceptable since the equipment remains under appropriate administrative control while it is in service prior to its OPERABILITY being formally established. The administrative controls will limit the time the equipment is in service to that which is necessary for performing the required testing. Furthermore, after corrective maintenance has been performed on the affected equipment, there is reasonable assurance that the required OPERABILITY testing will indeed demonstrate the equipment is capable of performing its safety function.
01-05	M	ITS LCO 3.0.6 as modified by TSTF-166 would be added to the CTS in conformance with NUREG-1431. ITS LCO 3.0.6 provides for an exception to LCO 3.0.2 for the case of an inoperable support system that has its own LCO specified in the Technical Specifications (TS.) When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if they are determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported system's ACTIONS unless directed to do so by the support system's Required ACTIONS. This exception is justified because the ACTIONS that are required to ensure the unit is maintained in a safe Condition are specified in the support system's LCO Required ACTIONS. Furthermore, the Safety Function Determination Program (SFDP) will ensure that there is no resulting loss of safety function. Finally, the potential confusion and inconsistency of requirements related to the entry into multiple support and supported system's LCO's ACTIONS are eliminated by providing all the actions that are necessary to ensure the unit is operated in an acceptably safe manner in the support system's Required Actions. While the proposed change clarifies CTS requirements regarding support system OPERABILITY, it also imposes the SFDP where no such program previously existed. Therefore, the proposed change is considered more restrictive.
01-06	A	New ITS LCO 3.0.7, as modified by Industry Travelers TSTF-12, Rev. 1 and TSTF-136, would be added to the CTS in conformance with NUREG-1431. This LCO provides for the optional application of test exception LCOs for the purpose of performing physics testing. The proposed change does not change the technical manner in which special test exceptions are administered; however, the addition of the LCO for this purpose is considered a change.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.0
(Continued)

CHANGE
NUMBER

NSHC

DESCRIPTION

01-07

LS3

CTS SR 4.0.2 is being changed in conformance with NUREG-1431 to apply the 25 percent extension to the Completion Times of repetitive Required Actions. This is a relaxation because it has been previously interpreted that the extension of SR 4.0.2 did not apply within ACTION statements. This change is considered acceptable as the current practice of performing the initial Required Action does not allow extension. Therefore, no loss of safety function is demonstrated within the required Completion Time. Following the initial performance, repetitive Required Actions are specifically delineated as SRs and as such it is appropriate for reasons of scheduling and plant Conditions that the 25 percent extension be allowable. As is the case with SRs, the extension is not intended to be used repeatedly merely as an operational convenience to extend periodic Completion Times beyond those specified.

01-08

LS4

CTS SR 4.0.3 is being changed with respect to the allowance for performing a missed surveillance upon discovery. This change is in conformance to NUREG-1431. The CTS allows the ACTION requirements to be delayed for up to 24 hours for completion of the surveillance when the allowed outage time (AOT) limits of the ACTION requirements are less than 24 hours after declaring the equipment inoperable, from the point of discovery. The proposed NUREG-1431 specification does not require the equipment to be declared inoperable and allows the lesser of 24 hours or the specified frequency for performance of the surveillance.

The allowance to not declare the equipment inoperable upon discovery of a missed surveillance is a relaxation in that ACTIONS for inoperable equipment are not entered solely due to a missed surveillance. The change in time to perform the surveillance could either be a relaxation or restriction based on whether the surveillance frequency or the AOT was more restrictive. The new requirement is based on time to perform a surveillance, and is therefore more consistent. The change still restricts performance of the surveillance to within 24 hours of discovery and is seen as acceptable, from the perspective of safety, as equipment is normally demonstrated OPERABLE, not inoperable via surveillance performance.

01-09

LS1

CTS SR 4.0.4 was previously applicable for entry into all MODES. The specification has been revised to not prevent unit shutdown and will apply only to entry into MODES 1, 2, 3, and 4 from lower MODES. This change is less restrictive in that it will allow MODE changes in either direction into MODES 5 and 6 prior to the performance of SRs. As required in the "Reviewer's Note" of NUREG-1431, SR 3.0.4, a matrix is provided (see LS1) which documents the plant-specific review of all specifications for determination of where specific restrictions on MODE changes or Required Actions should be included in individual LCOs. The change is acceptable for those specifications for which specific restrictions were not deemed warranted.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.0
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-10	A	TS 4.0.5, the SR for inservice testing, is moved to the ITS Administrative Controls Section 5.5.8 consistent with NUREG-1431. The reference to 10CFR50.55a is unnecessary and has been deleted.
01-11	LG	The portions of CTS 4.0.5 concerning inservice inspection (ISI) are moved to the ISI Program Plan. The requirements of 10CFR50.55a are adequate and TS are not necessary.
01-12	A	Consistent with NUREG-1431 Inservice Testing Program, ITS 5.5.8, the CTS is clarified to address the Inservice Testing (IST) Frequency (in days) for biennial requirements which was previously inferred but not explicitly stated in the CTS.
01-13	A	This change adds Applicability of CTS SR 4.0.3 (SR 3.0.3 in the ITS) for inservice testing for consistency with the wording in NUREG-1431. In NUREG-1431, inservice testing is moved to the Programs and Manuals Section (ITS 5.5.8) and is no longer a SR. Thus, an explicit statement that ITS SR 3.0.3 (CTS SR 4.0.3) was applicable to inservice testing was necessary to provide for the performance of missed IST requirements. CTS SR 4.0.5 (IST) is already an SR, and thus by definition, CTS SR 4.0.3 applies.
01-14	A	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-15	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-16	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-17	A	Consistent with the wording used in NUREG-1431, Section 5.5.8, CTS SR [4.0.5d] concerning performance of IST being performed in addition to other specified SRs, is deleted. The statement is redundant to the usage rules and is not necessary.
01-18	LS5	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-19	A	The phrase "Exceptions to this Specification are stated in the individual specifications" was added to CTS 4.0.2 to provide exceptions in individual frequencies where the allowances provided in CTS 4.0.2 are not allowed. This additional wording will have no impact on current practice regarding compliance with SRs. [The statement in CTS 4.0.3, "Exceptions to these requirements are stated in the individual specifications" is deleted in accordance with NUREG-1431.]
01-20	A	Consistent with NUREG-1431, the phrase "or variables outside specified limits" is added to the statement that surveillances do not have to be performed on inoperable equipment. This is not a technical change in that "variables outside specified limits" is now explicitly stated in the ITS; whereas, in the CTS, it was implicit in the definition of inoperable equipment.

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(3 pages)

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.0

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	Adds exceptions to current LCO 3.0.1 and LCO 3.0.2 as provided in proposed ITS LCOs 3.0.5, 3.0.6, and 3.0.7. []	Yes	Yes	Yes	Yes
01-02 LS1	CTS LCO 3.0.4 was previously applicable for entry into all MODES. The specification has been revised to not prevent unit shutdowns and will apply only to entry into MODES 1, 2, 3, and 4 from lower MODES.	Yes	Yes	Yes	Yes
01-03 A	LCO 3.0.5 [and SR 4.0.6 are] being deleted.	Yes	Yes	No, not in CTS.	No, not in CTS.
01-04 LS2	ITS LCO 3.0.5 is added which provides for returning to service under administrative control, equipment previously declared inoperable or removed from service, for the purpose of demonstrating its OPERABILITY or the OPERABILITY of other equipment.	Yes	Yes	Yes	Yes
01-05 M	LCO 3.0.6 is added which provides for an exception to LCO 3.0.2 for the case of an inoperable support system that has its own LCO specified in the TS.	Yes	Yes	Yes	Yes
01-06 A	LCO 3.0.7 is added which provides for the optional application of test exception LCO for the purpose of performing PHYSICS TESTING.	Yes	Yes	Yes	Yes
01-07 LS3	SR 4.0.2 is changed to apply the 25% extension to the Completion Times of repetitive required ACTIONS.	Yes	Yes	Yes	Yes
01-08 LS4	SR 4.0.3 is changed with respect to the allowance for performing a missed surveillance upon discovery.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.0

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-09 LS1	SR 4.0.4 was previously applicable for entry into all MODES. The specification has been revised to not prevent unit shutdowns and will apply only to entry into MODES 1, 2, 3, and 4 from lower MODES.	Yes	Yes	Yes	Yes
01-10 A	Specification 4.0.5, the SR for IST, are moved to the Administrative Controls Section 5.5.8.	Yes	Yes	Yes	Yes
01-11 LG	The portions of Specification 4.0.5 concerning ISI are moved to the ISI Program Plan.	Yes	Yes	Yes	Yes
01-12 A	This change adds clarification to address biennial requirements.	Yes	Yes	Yes	Yes
01-13 A	This change clarifies the Applicability of CTS SR 4.0.3 (SR 3.0.3 in ITS) for IST.	Yes	Yes	Yes	Yes
01-14 A	The SRs for the SGs are reformatted and moved to ITS SR 3.4.13.2 and the Administrative Controls Section 5.5.9.	No, SG surveillance is in CTS 3/4.4.5.	Yes	No, SG surveillance is in CTS 3/4.4.5.	No, SG surveillance is in CTS 3/4.4.5.
01-15 A	Clarification to remove potential interpretation problems related to probe orientation versus entry point.	No, SG surveillance is in CTS 3/4.4.5.	Yes	No, SG surveillance is in CTS 3/4.4.5.	No, SG surveillance is in CTS 3/4.4.5.
01-16 A	The reporting requirement for SG tube inspections is moved to improved TS 5.6.10.	No, SG surveillance is in CTS 3/4.4.5.	Yes	No, SG surveillance is in CTS 3/4.4.5.	No, SG surveillance is in CTS 3/4.4.5.
01-17 A	CTS SR [4.0.5.d] is deleted. Performance of other specified SRs is mandated in the individual TS and does not require a separate section to ensure compliance.	Yes	Yes	Yes	Yes
01-18 LS5	This change revises LCO 3.0.4 to allow MODE entry if associated ACTION has no time limit, consistent with NRC GL 87-09.	No, already in CTS.	No, already in CTS.	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-19 A	The phrase "Exceptions to this Specification are stated in the individual specifications" was added to CTS SR 4.0.2 to provide exceptions in individual Frequencies where the allowances provided in SR 4.0.2 are not allowed. [The statement in CTS 4.0.3, "Exceptions to these requirements are stated in the individual specifications" is deleted in accordance with NUREG-1431.]	Yes	No, already in CTS.	Yes	Yes
01-20 A	This phrase "or variables outside specified limits" is added to the statement that surveillances do not have to be performed on inoperable equipment.	Yes	Yes	Yes	Yes

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

PAGE

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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information may be moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3...." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

LCO 3.0.4 and SR 4.0.4 in the CTS are applicable for entry into all MODES. The proposed change revises the applicability of these specifications to not prevent unit shutdowns and will apply to entry into MODES 1, 2, 3, and 4 only. This change is less restrictive in that it will allow MODE changes from either direction into MODES 5 and 6 while operating within an ACTION with a finite time period or prior to performance of a SR. This change is also less restrictive in that MODE changes that are part of a unit shutdown are not prevented.

All specifications were evaluated for individual acceptability of this application. Based on this evaluation, (see attached LCO 3.0.4 evaluation and matrix), where MODE change restrictions were determined to be required in MODES 5 and 6, or in MODES 1, 2, 3, and 4 during unit shutdown, notes containing the appropriate MODE change restrictions were added to the individual specifications.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change has impact on what equipment is required to be OPERABLE or demonstrated OPERABLE via surveillance prior to unit shutdowns or entry into MODES 5 and 6. This change could increase the probability or consequences of an accident previously evaluated if applied without consideration to all applicable transitions. However, as part of the change, an evaluation is attached in the form of a matrix, that identifies those specifications which must continue to apply LCO 3.0.4 and SR 3.0.4. Therefore, only those specifications that do not impact safety for these plant conditions, are afforded this relaxation.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change administratively changes when equipment is required to be OPERABLE or demonstrated OPERABLE via surveillance prior to entry into MODES 5 and 6. However, the proposed change does not introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change administratively changes what equipment is required to be OPERABLE or demonstrated OPERABLE via surveillance prior to unit shutdown or entry into MODES 5 and 6. However, the proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS 1" resulting from the conversion to the ITS format, satisfy the NSHC standards of 10CFR50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2

10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS
WITHIN THE TECHNICAL SPECIFICATIONS

ITS LCO 3.0.5 provides for an allowance to return equipment which had previously been declared inoperable back to service to either demonstrate the equipment's OPERABILITY or the OPERABILITY of other equipment.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

ITS LCO 3.0.5 provides for an allowance to return equipment which had previously been declared inoperable back to service to either demonstrate the equipment's OPERABILITY or the OPERABILITY of other equipment. Returning inoperable equipment to service under administrative control for testing will be to demonstrate equipment OPERABILITY. The equipment remains under appropriate administrative control while it is in service prior to its OPERABILITY being formally established. The expected result is that the equipment will be restored to OPERABLE status. In any case, however, the situation of inoperable safety-related equipment is controlled within the limits provided in the Applicability and ACTION statements to ensure there are no significant changes to risks or consequences as a result of the process of returning equipment to service.

Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change administratively changes the allowance to return equipment to service when inoperable for demonstrating that the equipment is OPERABLE or other equipment is OPERABLE. However, the proposed change does not introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "LS 2" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10CFR50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

CTS SR 4.0.2 is being revised to allow application of the 25 percent extension for surveillance performance to the Completion Times of repetitive Required ACTIONS, following the initial performance. The extension is not allowed on initial performance as the first performance is demonstrating a necessary OPERABILITY. The subsequent performances are confirming that OPERABILITY is being maintained.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The extension is not applied to the initial performance which is required to demonstrate a component or system is OPERABLE, and therefore continued operation within the confines of the Condition are acceptable. The subsequent performances of the requirement are not time critical as the expectation is that the performance will confirm OPERABILITY. Therefore, there is no reduction in requirements for the initial performance and only up to a maximum of 25 percent extension (the same as for routine surveillances) on subsequent performance. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change relaxes the Completion Times for performing repetitive ACTIONS designed to confirm OPERABILITY within a Required Action. This change is consistent with the allowances for performing routine surveillances which are anticipated to confirm OPERABILITY. However, the proposed change does not introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "LS 3" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10CFR50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed change revises the time allowed to perform a missed surveillance upon discovery from up to 24 hours. When the AOT limits of the ACTION requirements are less than 24 hours, the AOT may be the lesser of 24 hours or the surveillance frequency interval. Additionally, the equipment is not required to be declared inoperable at the time of discovery of a missed surveillance. This prevents unnecessary entry into Conditions based solely on equipment inoperabilities when the equipment is anticipated to be demonstrated OPERABLE by surveillance performance. The change is acceptable because it still restricts performance of the surveillance to within 24 hours of discovery and the equipment is normally demonstrated to be OPERABLE, not inoperable, by performance of the surveillance.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Surveillance performance is nominally considered to demonstrate equipment OPERABILITY. Therefore, identifying that a surveillance has been missed usually does not result in any true inoperabilities existing as the equipment is normally demonstrated OPERABLE upon performance of the missed surveillance. This change potentially allows a longer time in some instances for performance of the surveillance from time of discovery that the surveillance was missed. In these cases, if the equipment were truly inoperable, there would be an extended duration in which the appropriate Required ACTIONS were not taken. Not taking the Required ACTIONS could have a negative effect on the probability or consequences of an accident. However, at no time will this period exceed 24 hours. Based on the short duration, a minimal impact if any, would be expected to overall plant safety. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change potentially relaxes the allowed times for performing a surveillance once identified as missed. However, the proposed change does not introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "LS4" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10CFR50.92(c), and accordingly a NSHC finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers (1 Page)

NUREG-1431 Specifications that are not applicable (1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
LCO 3.0.1	3.0-1
LCO 3.0.2	3.0-1
LCO 3.0.3	3.0-1
LCO 3.0.4	3.0-2
LCO 3.0.5	3.0-2
LCO 3.0.6	3.0-2
LCO 3.0.7	3.0-3
SR 3.0.1	3.0-4
SR 3.0.2	3.0-4
SR 3.0.3	3.0-4
SR 3.0.4	3.0-5

Methodology (2 Pages)

Industry Travelers Applicable to Section 3.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-06, Rev 1	Incorporated	3.0-01	
TSTF-08, Rev 2	Incorporated	N/A	
TSTF-12, Rev 1	Incorporated	3.0-02	
TSTF-52	Incorporated	N/A	
TSTF-71	Not Incorporated	N/A	Will be addressed in SFDP.
TSTF-103	Not Incorporated	N/A	Performed 3.0.4 Matrix.
TSTF-104	Incorporated	3.0-03	
TSTF-122	Not Incorporated	N/A	Not NRC approved as of the Traveler cutoff date.
TSTF-136	Incorporated	3.0-02	
TSTF-165	Incorporated	N/A	LCO 3.0.5 Bases change only.
TSTF-166	Incorporated	3.0-04	

NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

None

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and ~~LCO 3.0.7~~.

3.0.01

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

- a. MODE 3 within 7 hours;
- b. MODE 4 within 13 hours; and
- c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

(Continued)

3.0 LCO APPLICABILITY (Continued)

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This ~~LCO 3.0.4~~ Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit. ED

Exceptions to this Specification are stated in the individual Specifications. ~~These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.~~ 3:0:03

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

~~Reviewers's Note: LCO 3.0.4 has been revised so that changes in MODES or other specified conditions in the Applicability that are part of a shutdown of the unit shall not be prevented. In addition, LCO 3.0.4 has been revised so that it is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3. The MODE change restrictions in LCO 3.0.4 were previously applicable in all MODES. Before this version of LCO 3.0.4 can be implemented on a plant specific basis, the licensee must review the existing technical specifications to determine where specific restrictions on MODE changes or Required Actions should be included in individual LCOs to justify this change; such an evaluation should be summarized in a matrix of all existing LCOs to facilitate NRC staff review of a conversion to the STS.~~

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

(Continued)

3.0 LCO APPLICABILITY (Continued)

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, ~~additional evaluations and limitations may be required shall be performed~~ in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

3.0-04

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7 Test Exception LCOs ~~[3.1.8, 3.1.10, 3.1.11, and 3.4.19]~~ allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0-02

B-PS

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(Continued)

3.0 SR APPLICABILITY (Continued)

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions -** The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions -** The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications -** The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes-** Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts -** The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

Methodology For Mark-up of NUREG-1431 Specifications
(Continued)

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES CONTENTS

Mark-up:

<u>BASES</u>	<u>PAGE</u>
LCO 3.0.1	B 3.0-1
LCO 3.0.2	B 3.0-1
LCO 3.0.3	B 3.0-2
LCO 3.0.4	B 3.0-4
LCO 3.0.5	B 3.0-6
LCO 3.0.6	B 3.0-7
LCO 3.0.7	B 3.0-8
SR 3.0.1	B 3.0-10
SR 3.0.2	B 3.0-11
SR 3.0.3	B 3.0-12
SR 3.0.4	B 3.0-13
Methodology	(2 Pages)

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs LCO 3.0.1 through LCO 3.0.6 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1 LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2 LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

- a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and
- b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation

(Continued)

BASES

LCO 3.0.2 (Continued) Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3 LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met an

- a. An associated Required Action and Completion Time is not met and no other Condition applies; or

(Continued)

BASES

LCO 3.0.3
(Continued)

- b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that PERMITS routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

- a. The LCO is now met.
- b. A Condition exists for which the Required Actions have now been performed.
- c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited

(Continued)

BASES

LCO 3.0.3 (Continued) The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides ACTIONS for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.15, "Spent Fuel Storage Pool Water Level." LCO 3.7.15 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.15 are not met while in MODE 1, 2, or 3, or 4, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.15 of "Suspend movement of irradiated fuel assemblies in the spent fuel storage pool" is the appropriate Required Action to complete in lieu of the ACTIONS of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4 LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist

(Continued)

BASES

LCO 3.0.4
(Continued)

- a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and
- b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. ~~These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time.~~ Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from LCO 3.0.4 MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. In some cases (e.g., where a review has concluded that specific restriction on MODE changes should be included) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition

(Continued)

BASES

LCO 3.0.4
(Continued)

in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability. Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s) to allow the performance of SRs required testing to demonstrate:

- a. The OPERABILITY of the equipment being returned to service; or
- b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs - required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs - required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking

(Continued)

BASES

LCO 3.0.5 (Continued) an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR ~~required testing~~ on another channel in the same trip system.

LCO 3.0.6 LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2

(Continued)

BASES

LCO 3.0.6 (Continued) Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7 There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs ~~3.1.8, 3.1.10, and 3.1.11~~ and ~~3.4.19~~ allows specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1 SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

- a. The systems or components are known to be inoperable, although still meeting the SRs; or
- b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status

(Continued)

BASES

SR 3.0.1 (Continued) Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2 SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is ~~the Containment Leakage Rate Testing Program~~ a Surveillance with a Frequency of ~~"in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions."~~ The requirements of regulations take precedence over the TS. ~~The TS cannot in and of themselves extend a test interval specified in the regulations.~~

(Continued)

BASES

SR 3.0.2 ~~Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not~~
(Continued) ~~applicable."~~

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3 SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing

(Continued)

BASES

SR 3.0.3
(Continued)

the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or component to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

(Continued)

BASES

SR 3.0.4
(Continued)

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO ~~SR~~ 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4

(Continued)

BASES

SR 3.0.4 (Continued) The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

Methodology For Mark-up of NUREG-1431 Bases

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(1 Page)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.0

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE
NUMBER

JUSTIFICATION

- | | |
|--------|--|
| 3.0-01 | Industry Traveler TSTF-06, Rev.1, requests a PWR Generic (except CEOG) change to include LCO 3.0.7 as a specified exclusion to the Applicability of LCO 3.0.1. This exclusion is inherent in the discussion of LCO 3.0.7 and is added for consistency in TS interpretation. This change does not result in a technical change to application of the specifications. |
| 3.0-02 | Consistent with Traveler TSTF-12, Rev. 1, and TSTF-136, LCOs 3.1.9 and 3.1.11 are deleted and LCO 3.1.10 is relabeled to be LCO 3.1.8. The PHYSICS TESTS contained in LCO 3.1.9. were only contained in some initial plant startup testing programs. The PHYSICS TESTS can be deleted since these PHYSICS TESTS were never performed during past refueling outages. The PHYSIC TEST that LCO 3.1.11 required was the rod worth measurement in the N-1 Condition. The use of other Rod Worth Measurement techniques will maintain the SHUTDOWN MARGIN during the entire measurement process and still provide the necessary physics data verification. Since the N-1 measurement technique is no longer used, the SDM test exception can be deleted. ITS LCO 3.4.19 is not applicable because this test exception [is not in the DCPD CTS]. |
| 3.0-03 | This change provides consistency between LCO 3.0.4 and LCO 3.0.3 by moving the discussion of exceptions to LCO 3.0.4 to the Bases. This change is consistent with Industry Traveler TSTF-104. |
| 3.0-04 | This change provides consistency between LCO 3.0.6 and TS 5.5.15 and the Bases for LCO 3.0.6 to explicitly require SFDP evaluations. This change is consistent with Industry Traveler TSTF-166. |

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(1 Page)

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.0-01	Includes LCO 3.0.7 as a specified exclusion to the applicability of LCO 3.0.1.	Yes	Yes	Yes	Yes
3.0-02	Revises the bracketed test exception LCOs by deleting LCOs 3.1.9, 3.1.11, and 3.4.19 and re-labeling LCO 3.1.10 to LCO 3.1.8.	Yes	Yes	Yes	Yes
3.0-03	This change provides consistency between LCO 3.0.4 and LCO 3.0.3 by moving the discussion of exceptions to LCO 3.0.4 to the Bases. This change is consistent with Industry Traveler TSTF-104.	Yes	Yes	Yes	Yes
3.0-04	This change provides consistency between LCO 3.0.6 and TS 5.5.15 and the Bases for LCO 3.0.6 to explicitly require SFDP evaluations. This change is consistent with Industry Traveler TSTF-166.	Yes	Yes	Yes	Yes

JLS Conversion to Improved Technical Specifications Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 3/4.1 - Reactivity Control Systems

ITS 3.1 - Reactivity Control Systems



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.1

CONTENTS

- ENCLOSURE 1 - CROSS-REFERENCE TABLES
- ENCLOSURE 2 - MARK-UP OF CURRENT TS
- ENCLOSURE 3A - DESCRIPTION OF CHANGES TO CURRENT TS
- ENCLOSURE 3B - CONVERSION COMPARISON TABLE - CURRENT TS
- ENCLOSURE 4 - NO SIGNIFICANT HAZARDS CONSIDERATIONS
- ENCLOSURE 5A - MARK-UP OF NUREG-1431 SPECIFICATIONS
- ENCLOSURE 5B - MARK-UP OF NUREG-1431 BASES
- ENCLOSURE 6A - DIFFERENCES FROM NUREG-1431
- ENCLOSURE 6B - CONVERSION COMPARISON TABLE - NUREG 1431

ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(5 Pages)
CONVERSION TABLE SORTED BY IMPROVED TS	(4 Pages)
METHODOLOGY	(2 Pages)

CROSS-REFERENCE TABLE FOR 3/4.1
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.1.1.1	LCO		01-01-LG	3.1.1	LCO		3.1-1
3.1.1.1	LCO		02-01-A	3.1.2	LCO		
3.1.1.1	APP		01-06-A 02-01-A 03-01-A	3.1.1	APP		3.1-9
3.1.1.1	ACTION		01-03-LS1 01-07-LS16	3.1.1	CONDITION	A	
4.1.1.1.1	SR	a	01-04-LS2 12-03-A	3.1.4	CONDITION	A	3.1-6
4.1.1.1.1	SR	b	01-08-A	3.1.6.2	SR		
4.1.1.1.1	SR	c	01-09-A	3.1.6.1	SR		
4.1.1.1.1	SR	d	05-04-A	3.1.6.1	SR		
4.1.1.1.1	SR	e	01-05-LG 01-10-M	3.1.1.1	SR		3.1-1
4.1.1.1.2	SR		05-01-M	3.1.2.1	SR		3.1-2
4.1.1.1.2	SR		05-02-LS7	3.1.2.1	SR		
4.1.1.1.2	SR		05-03-LG 05-05-LS17	3.1.2.1	SR		
3.1.1.2	LCO		01-01-LG			Not Used	
3.1.1.2	APP		02-01-A	3.1.1	APP		3.1-9
3.1.1.2	ACTION		01-03-LS1			Not Used	
3.1.1.2	ACTION		01-07-LS16			Not Used	
4.1.1.2	SR	a	01-04-LS2 12-03-A			Not Used	
4.1.1.2	SR	b	01-05-LG			Not Used	
3.1.1.3	LCO			3.1.3	LCO		
3.1.1.3	LCO	a		3.1.3	Fig 3.1.4-1		
3.1.1.3	LCO	b	03-07-LG	3.1.3	LCO		
3.1.1.3	APP		03-01-A	3.1.3	APP		
3.1.1.3	ACTION	a.1	03-02-LS3	3.1.3	CONDITION	B	
3.1.1.3	ACTION	a.1	03-03-A	3.1.3	CONDITION	A	

CROSS-REFERENCE TABLE FOR 3/4.1
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.1.1.3	ACTION	a.2	03-04-LG			Not Used	
3.1.1.3	ACTION	a.3	03-05-TR-2			Not Used	
3.1.1.3	ACTION	b		3.1.3	CONDITION	C	
4.1.1.3	SR	a		3.1.3.1	SR		
4.1.1.3	SR	b		3.1.3.2	SR		3.1-4
4.1.1.3	SR	New	03-06-LS4	3.1.3.2	SR		3.1-4
3.1.1.4	LCO		04-03-A	3.4.2	LCO		
3.1.1.4	APP		03-01-A	3.4.2	APP		
3.1.1.4	ACTION		04-01-LS5	3.4.2	CONDITION	A	3.4-32
4.1.1.4	SR	a				Not Used	
4.1.1.4	SR	b	04-02-LS6	3.4.2.1	SR		3.4-33
3.1.1.5	New		01-06-A	3.1.2	LCO		
3.1.2.1	LCO		06-01-R			Not Used	
3.1.2.2	LCO		07-01-R			Not Used	
3.1.2.3	LCO		08-01-R			Not Used	
3.1.2.4	LCO		09-01-R			Not Used	
3.1.2.5	LCO		10-01-R			Not Used	
3.1.2.6	LCO		11-01-R			Not Used	
3.1.3.1	LCO		12-01-A 03-01-A	3.1.4	LCO		3.1-5 3.1-6
3.1.3.1	ACTION	a	12-06-A 12-14-LG	3.1.4	CONDITION	A	3.1-1 3.1-6

CROSS-REFERENCE TABLE FOR 3/4.1
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.1.3.1	ACTION	b	12-17-A	3.1.4	CONDITION	A	
3.1.3.1	ACTION	b.1	12-04-A	3.1.4	CONDITION	B.1	
3.1.3.1	ACTION	b.2	12-18-LG			Not Used	
3.1.3.1	ACTION	b.3	01-01-LG 12-04-A 12-06-A	3.1.4	CONDITION	B.2.1.1, B.2.1.2	3.1-1
3.1.3.1	ACTION	b.3.a	12-07-A	3.1.4	CONDITION	B.2.6	
3.1.3.1	ACTION	b.3.b	12-08-LS9 12-22-M	3.1.4	CONDITION	B.2.3	
3.1.3.1	ACTION	b.3.c		3.1.4	CONDITION	B.2.4	
3.1.3.1	ACTION	b.3.c		3.1.4	CONDITION	B.2.5	
3.1.3.1	ACTION	b.3.d	12-02-M	3.1.4	CONDITION	B.2.2	
3.1.3.1	ACTION	b.3.d		3.1.4	CONDITION	C	
3.1.3.1	ACTION	c	12-12-LS13			Not Used	
3.1.3.1	ACTION	d	12-02-M	3.1.4	CONDITION	D	3.1-1
Table 3.1-1			12-07-A	3.1.4	BASES		
4.1.3.1.1	SR		12-01-A 12-16-LG	3.1.4.1	SR		3.1-10
4.1.3.1.2	SR		12-01-A	3.1.4.2	SR		
4.1.3.1.3	SR		12-10-LS10	3.1.4.3	SR		
3.1.3.2	LCO		13-01-LG	3.1.7	LCO		
3.1.3.2	APP		13-08-LS20	3.1.7	APP		
3.1.3.2	ACTION	a	13-02-LS15	3.1.7	CONDITION	A	3.1-7,3.1-12
3.1.3.2	ACTION	a.1	13-03-LS12	3.1.7	CONDITION	A.1	3.1.-12
3.1.3.2	ACTION	a.2	13-04-A	3.1.7	CONDITION	A.2	3.1-12
3.1.3.2	ACTION	a.1		3.1.7	CONDITION	C	3.1-12
3.1.3.2	ACTION	NEW	13-08-LS20	3.1.7	CONDITION	NEW	3.1-7
3.1.3.2	ACTION	c	13-04-A	3.1.7	CONDITION	D	
3.1.3.2	ACTION	c	13-04-A	3.1.7	CONDITION	E	
4.1.3.2	SR		12-16-LG 13-07-M	3.1.7.1	SR		3.1-8

CROSS-REFERENCE TABLE FOR 3/4.1
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.1.3.3	LCO		03-01-A 14-01-R			Not Used	
4.1.3.3	SR		13-07-M	3.1.7.1	SR		3.1-8
3.1.3.4	LCO		15-01-R	3.1.4.3	SR		
3.1.3.4	LCO		12-10-LS10	3.1.4.3	SR		
3.1.3.4	ACTION		12-20-A	3.1.4.3	SR		
4.1.3.4	SR	a	15-02-A	3.1.4.3	SR		
4.1.3.4	SR	b, c	12-11-TR3			Not Used	
3.1.3.5	LCO		16-01-LS14	3.1.5	LCO		
3.1.3.5	APP		03-01-A 16-04-M	3.1.5	APP		
3.1.3.5	ACTION		16-01-LS14	3.1.5	CONDITION	A	
3.1.3.5	ACTION	a	16-02-M	3.1.5	CONDITION	A.2	
3.1.3.5	ACTION	b	01-01-LG 16-06-A	3.1.5	CONDITION	A.1.1, A.1.2	3.1-1
3.1.3.5	ACTION	c	16-05-M	3.1.5	CONDITION	B	
4.1.3.5	SR		16-01-LS14	3.1.5.1	SR		
4.1.3.5	SR	a	16-03-LS22			Not Used	
4.1.3.5	SR	b		3.1.5.1	SR		
3.1.3.6	LCO		17-01-M	3.1.6	LCO		
3.1.3.6	APP		03-01-A	3.1.6	APP		
3.1.3.6	ACTION	NEW	17-02-M	3.1.6	CONDITION	A.1.1, A.1.2	3.1-1
3.1.3.6	ACTION	NEW	17-02-M	3.1.6	CONDITION	B.1	
3.1.3.6	ACTION	b		3.1.6	CONDITION	A.2	
3.1.3.6	ACTION	b		3.1.6	CONDITION	B.2	
3.1.3.6	ACTION	c		3.1.6	CONDITION	A.2	
3.1.3.6	ACTION	d	17-03-LS21	3.1.6	CONDITION	C	

CROSS-REFERENCE TABLE FOR 3/4.1
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
4.1.3.6.1	SR		12-16-LG	3.1.6.2	SR		3.1-10
4.1.3.6.2	New		01-09-A	3.1.6.1	SR		
4.1.3.6.3	New		17-01-A	3.1.6.3	SR		

CROSS-REFERENCE TABLE FOR 3/4.1
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.1.1.1	LCO		01-01-LG	3.1.1	LCO		3.1-1
3.1.1.1	APP		01-06-A 02-01-A 03-01-A	3.1.1	APP		3.1-9
3.1.1.2	APP		02-01-A	3.1.1	APP		3.1-9
3.1.1.1	ACTION		01-03-LS1 01-07-LS16	3.1.1	CONDITION	A	
4.1.1.1.1	SR	e	01-05-LG 01-10-M	3.1.1.1	SR		3.1-1
3.1.1.1	LCO		02-01-A	3.1.2	LCO		
3.1.1.5	New		01-06-A	3.1.2	LCO		
3.1.1.1	APP		01-06-A	3.1.2	APP		
4.1.1.1.2	SR		05-02-LS7	3.1.2	CONDITION	A.1, A.2	
4.1.1.1.2	SR		05-05-LS17	3.1.2	CONDITION	B	
4.1.1.1.2	SR		05-01-M	3.1.2.1	SR		3.1-2
4.1.1.1.2	SR		05-02-LS7	3.1.2.1	SR		
4.1.1.1.2	SR		05-03-LG 05-05-LS17	3.1.2.1	SR		
3.1.1.3	LCO			3.1.3	LCO		
3.1.1.3	LCO	a		3.1.3	Fig 3.1.4-1		
3.1.1.3	LCO	b	03-07-LG	3.1.3	LCO		
3.1.1.3	APP		03-01-A	3.1.3	APP		
3.1.1.3	ACTION	a.1	03-03-A	3.1.3	CONDITION	A	
3.1.1.3	ACTION	a.1	03-02-LS3	3.1.3	CONDITION	B	
3.1.1.3	ACTION	b		3.1.3	CONDITION	C	
4.1.1.3	SR	a		3.1.3.1	SR		
4.1.1.3	SR	b		3.1.3.2	SR		3.1-4
4.1.1.3	SR	New	03-06-LS4	3.1.3.2	SR		3.1-4
3.1.3.1	LCO		12-01-A 03-01-A	3.1.4	LCO		3.1-5 3.1-6

CROSS-REFERENCE TABLE FOR 3/4.1
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.1.3.1	APP			3.1.4	APP		
4.1.1.1.1	SR	a	01-04-LS2 12-03-A	3.1.4	CONDITION	A	3.1-6
3.1.3.1	ACTION	a	12-06-A 12-14-LG	3.1.4	CONDITION	A	3.1-1 3.1-6
3.1.3.1	ACTION	b	12-17-A	3.1.4	CONDITION	A	
3.1.3.1	ACTION	b.1	12-04-A	3.1.4	CONDITION	B.1	
3.1.3.1	ACTION	b.3	01-01-LG 12-04-A 12-06-A	3.1.4	CONDITION	B.2.1.1, B.2.1.2	3.1-1
3.1.3.1	ACTION	b.3.a	12-07-A	3.1.4	CONDITION	B.2.6	
3.1.3.1	ACTION	b.3.b	12-08-LS9 12-22-M	3.1.4	CONDITION	B.2.3	3.1-1
3.1.3.1	ACTION	b.3.c		3.1.4	CONDITION	B.2.4	
3.1.3.1	ACTION	b.3.c		3.1.4	CONDITION	B.2.5	
3.1.3.1	ACTION	b.3.d	12-02-M	3.1.4	CONDITION	B.2.2	
3.1.3.1	ACTION	b.3.d		3.1.4	CONDITION	C	
3.1.3.1	ACTION	d	12-02-M	3.1.4	CONDITION	D	
4.1.3.1.1	SR		12-01-A 12-16-LG	3.1.4.1	SR		3.1-10
4.1.3.1.2	SR		12-01-A	3.1.4.2	SR		
4.1.3.1.3	SR		12-10-LS10	3.1.4.3	SR		
3.1.3.4	LCO		12-10-LS10 15-01-R	3.1.4.3	SR		
3.1.3.4	ACTION		12-20-A	3.1.4.3	SR		
4.1.3.4	SR		12-11-TR3 15-02-A	3.1.4.3	SR		
3.1.3.5	LCO		16-01-LS14	3.1.5	LCO		
3.1.3.5	APP		03-01-A 16-04-M	3.1.5	APP		
3.1.3.5	ACTION		16-01-LS14	3.1.5	CONDITION	A	
3.1.3.5	ACTION	b	01-01-LG 16-06-A	3.1.5	CONDITION	A.1.1, A.1.2	3.1-1

CROSS-REFERENCE TABLE FOR 3/4.1
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.1.3.5		A	16-02-M	3.1.5	CONDITION	A.2	
3.1.3.5	ACTION	c	16-05-M	3.1.5	CONDITION	B	
4.1.3.5	SR		16-01-LS14	3.1.5.1	SR		
4.1.3.5	SR	b		3.1.5.1	SR		
3.1.3.6	LCO		17-01-M	3.1.6	LCO		
3.1.3.6	APP		03-01-A	3.1.6	APP		
3.1.3.6	ACTION	NEW	17-02-M	3.1.6	CONDITION	A.1.1, A.1.2	3.1-1
3.1.3.6	ACTION	b		3.1.6	CONDITION	A.2	
3.1.3.6	ACTION	c		3.1.6	CONDITION	A.2	
3.1.3.6	ACTION	NEW	17-02-M	3.1.6	CONDITION	B.1	3.1-1
3.1.3.6	ACTION	b		3.1.6	CONDITION	B.2	
3.1.3.6	ACTION	d	17-03-LS21	3.1.6	CONDITION	C	
4.1.1.1.1	SR	c	01-09-A	3.1.6.1	SR		
4.1.3.6.2	New		01-09-A	3.1.6.1	SR		
4.1.3.6.1	SR		12-16-LG	3.1.6.2	SR		3.1-10
4.1.3.6.3	New		17-01-A	3.1.6.3	SR		
3.1.3.2	LCO		13-01-LG	3.1.7	LCO		
3.1.3.2	APP		13-08-LS20	3.1.7	APP		
3.1.3.2	ACTION	a	13-02-LS15 13-03-LS12 13-04-A	3.1.7	CONDITION	A	3.1-7 3.1-12
3.1.3.2	ACTION	NEW	13-08-LS20	3.1.7	CONDITION	NEW	3.1-7 3.1-12
3.1.3.2	ACTION	a		3.1.7	CONDITION	C	3.1-12
3.1.3.2	ACTION	c	13-04-A	3.1.7	CONDITION	D	
3.1.3.2	ACTION	c	13-04-A	3.1.7	CONDITION	E	
4.1.3.2	SR		12-16-LG 13-07-M	3.1.7.1	SR		3.1-8
4.1.3.3	SR		13-07-M	3.1.7.1	SR		3.1-8

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

Methodology for Cross-Reference Tables
(Continued)

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).
Note:	The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note: . When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE*</u>
3.1.1.1	3/4 1-1
3.1.1.2	3/4 1-3
3.1.1.3	3/4 1-4
3.1.1.4	3/4 1-6
3.1.2.1	3/4 1-7
3.1.2.2	3/4 1-8
3.1.2.3	3/4 1-10
3.1.2.4	3/4 1-11
3.1.2.5	3/4 1-12
3.1.2.6	3/4 1-13
3.1.3.1	3/4 1-15
3.1.3.2	3/4 1-18
3.1.3.3	3/4 1-19
3.1.3.4	3/4 1-20
3.1.3.5	3/4 1-21
3.1.3.6	3/4 1-22
Methodology	(2 Pages)

* Pages 3/4 1-9 through 3/4 1-16, 3/4 1-19, 3/4 1-21, and 3/4 1-22 are intentionally blank.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN — ~~T_{avg}~~ GREATER THAN 200°F

02-01-A

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% $\Delta k/k$ within the limits provided in the COER.

01-01-LG

APPLICABILITY: MODES 1, 2*, 3, and 4, and 5

02-01-A

ACTION:

01-06-A

With the SHUTDOWN MARGIN less than 1.6% $\Delta k/k$, immediately not within limit within 15 minutes initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or equivalent until the required to restore SHUTDOWN MARGIN is restored to within limit.

01-01-LG

01-07-LS16

01-03-LS1

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% $\Delta k/k$ within limits at least once per 24 hour:

01-01-LG

- a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);
- b. When in MODES 1 or 2 with K_{eff} greater than or equal to 1, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;
- c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;
- d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.e., below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

01-04-LS2

12-03-A

01-08-A

01-09-A

05-04-A

03-01-A

*See Special Test Exceptions Specification 3.10.1. With $K_{eff} \geq 1.0$

01-06-A

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 2, 3 or 4, at least once per 24 hours by consideration of the following factors: 01-10-M

1) ~~Reactor Coolant System boron concentration.~~ 01-05-LG

2) ~~Control rod position.~~

3) ~~Reactor Coolant System average temperature.~~

4) ~~Fuel burnup based on gross thermal energy generation.~~

5) ~~Xenon concentration, and~~

6) ~~Samarium concentration.~~

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm 1\% \Delta k/k$ once prior to entering MODE 1 after each refueling and at least once per 31 Effective Full Power Days (EFPD) after burnup > 60 EFPD. This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. If the reactivity balance is not within limits within 72 hours, evaluate the Safety Analyses and establish appropriate operating restrictions and/or surveillance requirements or be in at least MODE 3 within the next 6 hours. 05-01-M
05-03-LG
05-02-LS7
05-05-LS17

~~_____~~ 01-10-M

~~With $k_{eff} > 1.0$~~

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN — T_{avg} LESS THAN OR EQUAL TO 200°F

02-01-A

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 1% $\Delta k/k$ within the limits provided in the COLR.

01-01-LG

APPLICABILITY: MODE 5.

02-01-A

ACTION:

With the SHUTDOWN MARGIN less than 1% $\Delta k/k$, immediately not within limit, within 15 minutes initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored to within limit.

01-01-LG

01-07-LS16

01-03-LS1

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1% $\Delta k/k$ within limits at least once per 24 hours:

01-01-LG

- a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

01-04-LS2

12-03-A

- b. At least once per 24 hours by consideration of the following factors:

- 1) Reactor Coolant System boron concentration,
- 2) Control rod position,
- 3) Reactor Coolant System average temperature,
- 4) Fuel burnup based on gross thermal energy generation,
- 5) Xenon concentration, and
- 6) Samarium concentration.

01-05-LG

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be ~~within the limits specified in the COLR:~~ 03-07-LG

- a. ~~The maximum upper limit shall be less positive than $+5 \times 10^{-5} \Delta k/k/^\circ F$ for 0% to 70% RATED THERMAL POWER, and for $\geq 70\%$ to 100% RATED THERMAL POWER the MTC decreases linearly to $0 \Delta k/k/^\circ F$ for the all rods withdrawn condition, beginning of cycle life (BOL); or~~
- b. ~~Less negative than $-3.9 \times 10^{-4} \Delta k/k/^\circ F$ specified in the COLR for all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.~~ 03-07-LG

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#. 03-01-A
Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.

ACTION:

- a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:
 - 1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the limit of Specification 3.1.1.3a within 24 hours or be in ~~HOT STANDBY MODE 2 with $K_{eff} < 1.0$~~ in the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6; 03-02-LS3
03-03-A
 - 2. ~~The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and~~ 03-04-LG
 - 3. ~~A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.~~ 03-05-TR2
- b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.
- c. The provisions of Specification 3.0.4 are not applicable.

*With K_{eff} greater than or equal to 1.
#See ~~Special Test Exceptions Specification 3.10.3.~~

03-01-A

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

- a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and
- b. The MTC shall be measured at any THERMAL POWER and compared to $-3 \times 10^{-4} \Delta k/k/^\circ F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than $-3 \times 10^{-4} \Delta k/k/^\circ F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

(new) SR 4.1.1.3b need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of 60 ppm is less negative than the 60 ppm surveillance limit specified of the COLR

03-06-LS4

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

04-03-A

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 541°F.

APPLICABILITY: MODES 1 and 2#*.

03-01-A

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes ~~be in MODE 2 with $k_{eff} < 1.0$ within 30 minutes.~~

04-01-LS5

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F: ~~every 12 hours~~

- a. ~~Within 15 minutes prior to achieving reactor criticality, and~~
- b. ~~At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 551°F, with the $T_{avg} - T_{set}$ Deviation Alarm not reset.~~

04-02-LS6

#With K_{eff} greater than or equal to 1.

*See ~~Special Test Exceptions Specification 3.10.3.~~

03-01-A

LIMITING CONDITION FOR OPERATION

3.1.1.5 The measured core reactivity shall be within $\pm 1\%$ $\Delta k/k$ of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	72 hours <u>05-05-LS17</u>
	<u>AND</u> A.2 Establish appropriate operating restrictions and SRs.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>(new) -----NOTE----- The predicted reactivity values shall be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. ----- Verify measured core reactivity is within $\pm 1\% \Delta k/k$ of predicted values.</p>	<p><u>01-06-A</u></p> <p><u>05-03-LG</u></p> <p><u>05-04-A</u></p> <p><u>05-01-M</u></p> <p><u>05-02-LS7</u></p> <p>Once prior to entering MODE 1 after each refueling</p> <p><u>AND</u></p> <p>-----NOTE----- Only required after 60 EFPD ----- 31 EFPD thereafter</p>

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATH SHUTDOWN

06-01-R

LIMITING CONDITION FOR OPERATION

~~3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE with motor operated valves required to change position and pumps required to operate for boron injection capable of being powered from an OPERABLE emergency power source:~~

- ~~a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a. is OPERABLE, or~~
- ~~b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. is OPERABLE.~~

~~APPLICABILITY: MODES 5 and 6.~~

ACTION:

~~With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.~~

SURVEILLANCE REQUIREMENTS

~~4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:~~

- ~~a. At least once per 7 days by verifying that the temperature of the flow path is greater than or equal to 65°F when a flow path from the boric acid tanks is used, and~~
- ~~b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.~~

REACTIVITY CONTROL SYSTEMS

FLOW PATHS OPERATING

07-01-R

LIMITING CONDITION FOR OPERATION

~~3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:-~~

- ~~a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and~~
- ~~b. The flow path from the refueling water storage tank via a charging pump to the RCS.~~

~~APPLICABILITY: MODES 1, 2, 3 and 4#.~~

ACTION:-

- ~~a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.~~
- ~~b. With the flow path from the refueling water storage tank inoperable, restore the flow path to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

SURVEILLANCE REQUIREMENTS

~~4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:-~~

- ~~a. At least once per 7 days by verifying that the temperature of the flow path from the boric acid tanks is greater than or equal to 65°F.~~
- ~~b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.~~
- ~~c. At least once per 18 months by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal, and~~
- ~~d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a delivers at least 30 gpm to the RCS.~~

~~#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F.~~

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REACTIVITY CONTROL SYSTEMS

CHARGING PUMP SHUTDOWN

08-01-R

LIMITING CONDITION FOR OPERATION

~~3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.~~

~~APPLICABILITY: MODES 5 and 6.~~

ACTION:

~~With no charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.~~

SURVEILLANCE REQUIREMENTS

~~4.1.2.3.1 At least the above required charging pump shall be demonstrated OPERABLE when tested pursuant to Specification 4.0.5. In addition, when the above required charging pump is a centrifugal charging pump, verify that, on recirculation flow, the centrifugal charging pump develops a differential pressure of greater than or equal to 2400 psid.~~

~~4.1.2.3.2 All centrifugal charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable* at least once per 12 hours, except when the reactor vessel head is removed, by verifying that the motor breaker D.C. control power is de-energized.~~

~~*An inoperable pump may be made OPERABLE for testing per Specification 4.0.5 provided the discharge of the pump has been isolated from the Reactor Coolant System by an isolation valve with power removed from the valve operator, or by a sealed closed manual isolation valve.~~

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS OPERATING

09-01-R

LIMITING CONDITION FOR OPERATION

~~3.1.2.4 At least two charging pumps shall be OPERABLE.~~

~~APPLICABILITY: MODES 1, 2, 3 and 4#.~~

ACTION:

~~With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.~~

SURVEILLANCE REQUIREMENTS

~~4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE when tested pursuant to Specification 4.0.5. In addition, when the above required charging pumps include a centrifugal charging pump(s), verify that, on recirculation flow, each required centrifugal charging pump(s) develops a differential pressure of greater than or equal to 2400 psid.~~

~~4.1.2.4.2 All centrifugal charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable* at least once per 12 hours whenever the temperature of one or more of the Reactor Coolant System (RCS) cold legs is less than or equal to 270°F by verifying that the motor breaker D.C. control power is de-energized.~~

~~#A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F.~~

~~*An inoperable pump may be made OPERABLE for testing per Specification 4.0.5 provided the discharge of the pump has been isolated from the Reactor Coolant System by an isolation valve with power removed from the valve operator, or by a sealed closed manual isolation valve.~~

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCE SHUTDOWN

10-01-R

LIMITING CONDITION FOR OPERATION

~~3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:~~

- ~~a. A Boric Acid Storage System with:
 - ~~1) A minimum contained borated water volume of 2,499 gallons,~~
 - ~~2) A boron concentration between 7,000 and 7,700 ppm, and~~
 - ~~3) A minimum solution temperature of 65°F.~~~~
- ~~b. The Refueling Water Storage Tank (RWST) with:
 - ~~1) A minimum contained borated water volume of 50,000 gallons,~~
 - ~~2) A minimum boron concentration of 2300 ppm, and~~
 - ~~3) A minimum solution temperature of 35°F.~~~~

~~APPLICABILITY: MODES 5 and 6.~~

ACTION:

~~With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.~~

SURVEILLANCE REQUIREMENTS

~~4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:~~

- ~~a. At least once per 7 days by:
 - ~~1) Verifying the boron concentration of the water,~~
 - ~~2) Verifying the contained borated water volume, and~~
 - ~~3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.~~~~
- ~~b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside ambient air temperature is less than 35°F.~~

LIMITING CONDITION FOR OPERATION

~~3.1.2.6 The following borated water source shall be OPERABLE:~~

- ~~a. A Boric Acid Storage System with:~~
 - ~~1) A minimum contained borated water volume of 14,042 gallons,~~
 - ~~2) A boron concentration between 7,000 and 7,700 ppm, and~~
 - ~~3) A minimum solution temperature of 65°F.~~

~~APPLICABILITY: MODES 1, 2, 3 and 4.~~

ACTION:

- ~~a. With the Boric Acid Storage System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.~~

SURVEILLANCE REQUIREMENTS

~~4.1.2.6 The borated water source shall be demonstrated OPERABLE:~~

~~a. At least once per 7 days by:~~

~~1) Verifying the boron concentration in the water,~~

~~2) Verifying the contained borated water volume of the water source,
and~~

~~3) Verifying the Boric Acid Storage System solution temperature.~~

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group demand position.

12-01-A

APPLICABILITY: MODES 1* and 2*.

03-01-A

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable inoperable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied is within the limits in the COLR or initiate boration to restore the SHUTDOWN MARGIN within limits within 1 hour and be in HOT STANDBY within 6 hours.

12-01-A

12-14-M

01-01-LG

12-06-A

b. With one full length rod trippable but inoperable due to causes other than addressed by ACTION a. above, or misaligned from its group demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

12-01-A

12-17-A

1. The rod is restored to OPERABLE status within the above alignment requirements, or

12-04-A

2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable misaligned rod are aligned to within ± 12 steps of the inoperable misaligned rod or while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

12-18-LG

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied is verified to be within the limits provided in the COLR or initiate boration to restore the SHUTDOWN MARGIN within limits. POWER OPERATION may then continue provided that:

12-04-A

01-01-LG

12-06-A

a) A reevaluation of each accident analysis of Table 3.1.1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions; and

12-07-A

b) THE SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined is verified to be within limits at least once per 12 hours;

01-01-LG

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

- c) A power distribution map is obtained from the movable incore detectors and $F_0(Z)$ and F_{th} are verified to be within their limits within 72 hours; and
- d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER or be in HOT STANDBY within 6 hours 12-08-LS9
12-22-M
- c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a above, POWER OPERATION may continue provided that: 12-12-LS13
 - 1. Within 1 hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and
 - 2. The inoperable rods are restored to OPERABLE status within 72 hours.
- d. With more than one rod misaligned from its group demand position by more than $+ 12$ steps (indicated position), determine that the SHUTDOWN MARGIN is within the COLR or initiate boration to restore the SHUTDOWN MARGIN within limits within one hour, and be in HOT STANDBY within 6 hours. 01-01-LG
12-02-M

SURVEILLANCE REQUIREMENTS

- 4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours. 12-01-A
12-16-LG
- 4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 92 days. 12-01-A
- 4.1.3.1.3 The rod drop time of rods shall be demonstrated less than or equal to 2.7 seconds from beginning of decay of the stationary gripper coil voltage to dash pot entry with t_{rod} greater than or equal to 500 °F and all reactor coolant pumps operating, prior to criticality after each reactor head removal. 12-10-LS10

TABLE 3.1.1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD

~~Rod Cluster Control Assembly Insertion Characteristics~~

12-07-A

~~Rod Cluster Control Assembly Misalignment~~

~~Loss Of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System~~

~~Single Rod Cluster Control Assembly Withdrawal At Full Power~~

~~Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)~~

~~Major Secondary System Pipe Rupture~~

~~Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)~~

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

13-01-LG

APPLICABILITY: MODES 1* and 2*.

13-08-LS20

ACTION:

- a. With a maximum of one digital rod position indicator per bank group inoperable for one or more groups either:
- 13-02-LS15
- Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately within 4 hours after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

13-03-LS12

 - Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours or be in Hot Standby within the next 6 hours.

13-04-M
- b. With more than one digital rod position indicator per group inoperable either:
- a) Determine the position of the nonindicating rods indirectly by the movable incore detectors at least once per 8 hours and within 4 hours after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, and

13-08-LS20

 - b) Restore the digital rod position indicators to OPERABLE status within 24 hours such that a maximum of one digital rod position indicator per group is inoperable, or

13-08-LS20

 - Be in HOT STANDBY within the next 6 hours.

13-08-LS20
- b c. With a maximum of one demand position indicator per bank inoperable either:
- Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or
 - Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours or be in Hot Standby within the next 6 hours.

13-04-M

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours when exercised over the full range of rod travel once prior to criticality after each removal of the reactor head.

13-07-M

12-16-LG

* Separate condition entry is allowed for each inoperable rod position indicator and each demand position indicator per bank.

13-08-LS20

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM SHUTDOWN

14-01-R

LIMITING CONDITION FOR OPERATION

~~3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within \pm 12 steps for each shutdown or control rod not fully inserted.~~

~~APPLICABILITY: MODES 3*#, 4*# and 5*#.~~

ACTION:

~~With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.~~

SURVEILLANCE REQUIREMENTS

~~4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full range of rod travel at least once per 18 months prior to criticality after each removal of the reactor head.~~

13-07-M

~~*With the Reactor Trip System breakers in the closed position.
#See Special Test Exceptions Specification 3.10.4~~

14-01-R
03-01-A

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

15-01-R

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

- a. Tavg greater than or equal to 541 ~~500~~°F, and
- b. All reactor coolant pumps operating.

12-10-LS10

APPLICABILITY: ~~MODES 1 and 2.~~

ACTION:

12-01-A

With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to ~~MODE 1 or 2 reactor criticality.~~

12-20-A

SURVEILLANCE REQUIREMENTS

12-01-A

4.1.3.4 The rod drop time of ~~full-length rods~~ shall be demonstrated through measurement prior to reactor criticality:

- a. For all rods following each removal of the reactor vessel head. 15-02-A
- b. ~~For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and~~ 12-11-TR3
- c. ~~At least once per 18 months.~~

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All ~~Each~~ shutdown rods bank shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

16-01-LS14

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a ~~maximum of one or more~~ shutdown rod banks inserted beyond the insertion limit specified in the COLR, ~~except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:~~

16-01-LS14

a. ~~Within 2 hours Restore the rod to Shutdown Banks within the insertion limits specified in the COLR, or and~~

16-02-M

b. ~~Declare the rod to be inoperable and apply Specification 3.1.3.1. Within 1 hour verify SHUTDOWN MARGIN to be within the limits provided in the COLR, or initiate boration to restore SHUTDOWN MARGIN to within limits, or~~

16-06-A

01-01-LG

c. ~~Be in Hot Standby within 6 hours~~

16-05-M

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod bank shall be determined to be within the insertion limits specified in the COLR ~~at least once per 12 hours~~

16-01-LS14

a. ~~Within 15 minutes prior to withdrawal of any rods in Control Banks A, B, C or D during an approach to reactor criticality, and~~

16-03-LS22

b. ~~At least once per 12 hours thereafter.~~

* ~~See Special Test Exceptions Specifications 3.10.2 and 3.10.3 Not applicable while performing SR 4.1.3.1.2.~~

03-01-A

~~With Keff greater than or equal to 1 With any Control Bank not fully inserted.~~

16-04-M

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion sequence and over lap limits as specified in the CORE OPERATING LIMITS REPORT (COLR).

17-01-M

APPLICABILITY: MODES 1* and 2*# except for surveillance testing pursuant to Specification 4.1.3.1.2.

03-01-A

ACTION:

With the control banks inserted beyond the insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2 or not within sequence and over lap:

17-01-M

- a. Within 1 hour, verify that the SHUTDOWN MARGIN is within the limits provided in the COLR or initiate boration until the SHUTDOWN MARGIN is restored to within limits.
- b. Restore the control banks to within the limits within 2 hours, or
- c. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the insertion limits specified in the COLR, or
- d. Be in at least HOT STANDBY within next 6 hours.

17-02-M

17-03-LS21

SURVEILLANCE REQUIREMENTS

4.1.3.6.1 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the rod insertion limit monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

12-16-LG

4.1.3.6.2 When in MODE 2, with K_{eff} less than 1, verify that the predicted critical Control Rod position is within insertion limits within 4 hours prior to achieving reactor criticality.

01-09-A

4.1.3.6.3 Verify sequence and over lap limits specified in the COLR are met for Control Banks not fully withdrawn from the core at least once per 12 hours.

17-01-M

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.

03-01-A

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. Deletions - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. Additions - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. Modifications - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. Administrative - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

Methodology For Mark-Up of Current TS
(Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers	(1 Page)
Description of Changes	(12 Pages)

TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS

SECTION 3/4.1

TECHNICAL SPECIFICATION TITLE	CHG. NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Shutdown Margin MODES 3, 4, & 5	01	None	3.1.1.1	None	None
Shutdown Margin MODES 3 & 4	01	3.1.1.1	None	None	None
Shutdown Margin (G.T. 200°F) MODE 1, 2, 3 & 4	01	None	None	3.1.1.1	3.1.1.1
Shutdown Margin (L.T.E. 200°F) MODE 5	02	3.1.1.2	None	3.1.1.2	3.1.1.2
Moderator Temp. Coefficient	03-	3.1.1.3	3.1.1.3	3.1.1.3	3.1.1.3
Figure 3.1-1, MTC vs RTP	03-	None	Figure 3.1-1	None	None
Min. Temperature for Criticality	04	3.1.1.4	3.1.1.4	3.1.1.4	3.1.1.4
Core Reactivity	05	3.1.1.5	3.1.1.5	None	None
Boration Path Shutdown	06	None	None	3.1.2.1	3.1.2.1
Boration Path Operating	07	None	None	3.1.2.2	3.1.2.2
Charging Pumps Shutdown	08	None	None	3.1.2.3	3.1.2.3
Charging Pumps Operating	09	None	None	3.1.2.4	3.1.2.4
Borated Water Sources Shutdown	10	None	None	3.1.2.5	3.1.2.5
Borated Water Sources Operating	11	None	None	3.1.2.6	3.1.2.6
Movable Control Assemblies - Group Height	12	3.1.3.1	3.1.3.1	3.1.3.1	3.1.3.1
Table 3.1-1	12	Tbl. 3.1-1	Tbl. 3.1-1	Tbl. 3.1-1	Tbl. 3.1-1
Position Indication - Operating	13	3.1.3.2	3.1.3.2	3.1.3.2	3.1.3.2
Position Indication - Shutdown	14	None	None	3.1.3.3	3.1.3.3
Rod Drop Time	15	None	None	3.1.3.4	3.1.3.4
Shutdown Rod Insertion Limits	16	3.1.3.5	3.1.3.5	3.1.3.5	3.1.3.5
Control Rod Insertion Limits	17	3.1.3.6	3.1.3.6	3.1.3.6	3.1.3.6

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	LG	In accordance with TSTF-9, Rev. 1, this change would move the specified limit for SHUTDOWN MARGIN (SDM) from Technical Specification (TS) to the CORE OPERATING LIMITS REPORT (COLR). This change occurs in several specifications including the specifications for SDM and those specifications with ACTIONS that require verifying SDM within limits. SDM is a cycle-specific parameter that is calculated based on an NRC approved methodology. Moving the SDM to the COLR will provide core design and operational flexibility that can be used for improved fuel management.
01-02	M	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-03	LS1	The ACTION statement would be modified to reflect that the requirement to initiate boration at a specified rate with fluid at a specified boron concentration is generalized to simply require boration. As described in the ITS Bases, the required flow rate and boron concentration should be selected depending on plant conditions and available equipment. The ITS Bases allow the operator to use the "best source available for the plant conditions." This is an example of maintaining the overall safety requirement in the TS, but removing procedural details from the TS allowing the plant operator the ability to select the appropriate procedure and equipment for the existing plant condition.
01-04	LS2	<p>The existing Surveillance Requirements (SRs) require SDM to be verified within 1 hour in the event of detecting inoperable control rods. Since the SRs apply to inoperable (untripable) rods, they should only be applicable in MODES 1 and 2 []. The control rods are used to maintain SDM in MODES 1 and 2 by maintaining the rods within insertion limits. The requirements for rod alignment limits in CTS specify ACTIONS to be taken upon detecting an inoperable rod(s). The ACTIONS include verifying SDM within 1 hour. Deletion of the CTS surveillance(s) to verify SDM when a rod(s) is inoperable in the shutdown in MODES [3 through 5] accomplishes the following:</p> <ol style="list-style-type: none">1) Deletes an inappropriate Applicability (i.e., the surveillance(s) should not apply in those MODES when the rods are not required to be OPERABLE); and,2) Deletes redundant requirements (i.e., the requirements are properly and fully addressed in the specifications related to rod alignment/OPERABILITY and insertion limits).

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-05	LG	The list of specific items to be considered in the performance of an SDM verification would be deleted. These items are listed in the ITS Bases. This change is of the type that removes unnecessary procedural details from the specifications while leaving the overall safety requirement intact.
01-06	A	This change revises the SDM limiting condition of operation (LCO) Applicability to MODE 2 with $k_{eff} < 1.0$, MODE 3, and MODE 4. This change also creates a new core reactivity LCO based on ITS 3.1.3. This is consistent with NUREG-1431.
01-07	LS16	The term "immediately" is changed to "15 minutes" which is more specific. The term "immediately" simply specifies a prompt ACTION. The term "completion time of 15 minutes" is meant to clearly state a completed ACTION. The requirements of this ACTION are met only if boron is already being injected at 15 minutes. This time period provides adequate time for the operator to align and start the required systems. This is consistent with NUREG-1431.
01-08	A	The technical contents of this SR (verification of SDM through compliance with rod insertion limits) in MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ have been incorporated into LCO 3.1.6 of the ITS.
01-09	A	The SR for verification of the estimated critical Condition during the approach to criticality is moved to ITS SR 3.1.6.1.
01-10	M	CTS SR 4.1.1.1.1.e is modified by this change to be applicable to MODE 2 with $k_{eff} < 1.0$ as well as current MODES 3 and 4. This is consistent with NUREG-1431.
02-01	A	In the conversion process, this LCO will be combined with the SDM LCO applicable for $T_{avg} > 200^\circ\text{F}$, in accordance with Traveler TSTF-136. Traveler TSTF-9, Rev. 1, relocated values for SDM to the COLR which removed the only difference between ITS LCO 3.1.1 and ITS LCO 3.1.2. Differences above and below 200°F will be addressed in the COLR.
03-01	A	The footnote referring to CTS special test exceptions would be deleted. This is acceptable because the requirements for special test exceptions are provided in separate LCOs. Therefore, a separate reference in the footnote is redundant.
03-02	LS3	ACTION Statement A.1 would be revised to require achieving MODE 2 with $k_{eff} < 1.0$ instead of achieving HOT STANDBY if the beginning of life (BOL) moderator temperature coefficient (MTC) limit is exceeded and revised rod withdrawal limits cannot be established. This change corrects a discrepancy between the BOL Applicability and the ACTION, while ensuring that the plant is taken to a Condition in which the LCO is not applicable. Revising the CTS, even to correct an inconsistency, represents a relaxation in ACTION Statement A.1.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
03-03	A	The statement that administrative withdrawal limits required to meet ACTION Statement a.1 are in addition to insertion limits of another specification would be removed. This change is an administrative change because the statement is redundant to the requirements of CTS 3.1.3.6 and, therefore, can be deleted.
03-04	LG	The requirement of CTS ACTION Statement a.2, which provides the details of how to verify that MTC has been restored to within limits (i.e., calculation) for the all rods withdrawn Condition prior to exiting ACTION Statement a.1, is addressed in the ITS 3.1.3 Bases.
03-05	TR2	The requirement to submit a special report to the NRC would be deleted. This deletion is consistent with NUREG-1431.
03-06	LS4	This change would incorporate a note from ITS 3.1.3, allowing suspension of MTC testing near the end of the cycle when further significant changes to the MTC would not occur and result in exceeding the end of life (EOL) limit. This represents a relaxation in performing the SR.
03-07	LG	This change will relocate the negative EOL moderator temperature limit to the COLR. This is consistent with NUREG-1431.
04-01	LS5	This proposed change would make two changes to the ACTION statement. First, it would alter the ACTION statement shutdown requirement time limit from a combination of 15 minutes to restore T_{avg} to within limits followed by 15 minutes to be in MODE 3, if T_{avg} could not be restored to a single 30 minute limit to exit the Applicability if T_{avg} were not within its limit. Second, the ACTION statement would be revised to require achieving MODE 2 with $k_{eff} < 1.0$ instead of achieving HOT STANDBY if the LCO were not met (refer to TSTF-26). Regarding the first change, both the CTS requirement and the ITS requirement are essentially equivalent in that the plant is now required to shutdown and exit the Applicability of the specification within 30 minutes after discovering that a parameter is not within its limits if the parameter is not restored to within its limits in that 30 minute time period. If the LCO can be satisfied at any time during the 30 minute time frame, the plant shutdown can be terminated. Regarding the second change, it represents a relaxation in current ACTION statement requirements for plant shutdown, consistent with exiting the LCO's Applicability.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1

(Continued)

CHANGE NUMBER

NSHC

DESCRIPTION

04-02

LS6

The proposed change would revise the conditional SR for verifying that reactor coolant system (RCS) temperature (T_{avg}) is within limits by changing the frequency to once per 12 hours in accordance with TSTF-27, Rev. 2. The original frequency requirements were within 15 minutes prior to achieving reactor criticality and at least once per 30 minutes when the reactor is critical and the ($T_{avg} - T_{ret}$) deviation alarm is not reset. The RCS temperature is maintained within limit: (1) to assure that the MTC is within the limits assured in the accident analysis, (2) to assure that the neutron detectors are not adversely affected by attenuation caused by low RCS temperature, (3) to assure that the RCS and pressurizer response to thermal hydraulic transients is as predicted, and (4) to assure that the reactor vessel temperature is above the nil-ductility transition reference temperature.

The plant design incorporates monitoring of T_{avg} and provides an alarm, the ($T_{avg} - T_{ret}$) deviation alarm, as T_{avg} approaches its limit. This alarm Condition requires a response by the operating staff. Therefore, at any time that T_{avg} is approaching its limiting value, the plant operators would receive an alarm and initiate corrective ACTION.

04-03

A

The LCO for Minimum Temperature For Criticality, CTS 3/4.1.1.4, is moved to ITS 3.4.2 in the RCS section. This is consistent with NUREG-1431.

05-01

M

The CTS SR that requires a comparison of measured reactivity with predicted would be modified to add the requirement for performance prior to entry into MODE 1 after refueling outages. This is a new requirement from the ITS that is not in CTS. The CTS Bases indicate that the comparison should be made at RATED THERMAL POWER (RTP) after startup from a refueling outage.

05-02

LS7

This proposed change would specify that the overall core reactivity balance comparison shall be done every 31 effective full power days (EFPD) after burnup exceeds 60 EFPD. This clarifies when the 31 EFPD surveillance should start. The current SR specifies a frequency of 31 EFPD. The delay in initiating the monthly surveillance is acceptable because of the slow rate of changes in the core due to fuel depletion and the presence of other indicators for prompt determination of an anomaly. As noted in the Bases for Specification 3.1.2, the reactivity balance comparison must be done within the first 60 EFPDs after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

05-03

LG

The list of specific items to be considered in the performance of a reactivity balance verification is moved to the Bases for ITS 3.1.2. This change is of the type that moves unnecessary details from the specifications while leaving the overall safety requirement intact. See also CN 01-05-LG.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1

(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
05-04	A	CTS SR requires a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits. SDM in MODES 1 and 2 is determined by shutdown and control rods maintained at their insertion limits. The relevant requirements regarding the adequacy of the SDM with rods at their insertion limits is determined through compliance with ITS 3.1.2, which requires a reactivity balance prior to entering MODE 1 after each refueling; and ITS SR 3.1.6.1, which requires a verification of control bank position within insertion limits within 4 hours prior to criticality. Therefore, the requirements of this SR would be performed by other specifications in the ITS. []
05-05	LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown. This is consistent with NUREG-1431.
05-06	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
06-01	R	The CTS 3.1.2.1, "Boration Flow Path Shutdown," and associated SR 4.1.2.1 are relocated outside of the TS. This is consistent with NUREG-1431.
07-01	R	The CTS 3.1.2.2, "Boration Flow Path Operating," and associated SR 4.1.2.2 are relocated outside of the TS. This is consistent with NUREG-1431.
07-02	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
08-01	R	The CTS 3.1.2.3, "Charging Pump Shutdown," and associated SRs 4.1.2.3.1 and 4.1.2.3.2 are relocated outside of the TS. This is consistent with NUREG-1431.
08-02	M	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
08-03	LS19	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
09-01	R	The CTS 3.1.2.4, "Charging Pump Operating," and associated SRs 4.1.2.4.1 and 4.1.2.4.2 are relocated outside of the TS. This is consistent with NUREG-1431.
10-01	R	The CTS 3.1.2.5, "Borated Water Source Shutdown," and associated CTS SR 4.1.2.5 are relocated outside of the TS. This is consistent with NUREG-1431.
11-01	R	The CTS 3.1.2.6, "Borated Water Source Operating," and associated CTS SR 4.1.2.6 are relocated outside of the TS. This is consistent with NUREG-1431.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
12-01	A	The term "full-length" has been deleted. The plant design does not include full- and part-length control rods. Also, the LCO defines the indicated position limits that must be maintained.
12-02	M	Consistent with NUREG-1431, the ACTIONS required for more than one misaligned, but operable rod would be changed to be identical to those for inoperable rod(s). The current ACTION b. requires achieving HOT STANDBY within 6 hours. The ITS ACTIONS would require SDM verification and restoration as well as achieving HOT STANDBY within 6 hours. Therefore, adopting the ITS requirement would add additional restrictions to assure safe plant operation.
12-03	A	The requirement to include an increased allowance in the SDM calculation for the untrippable rod is addressed by the definition of SDM. The ITS 3.1.4 Bases for ACTION A.1.1 and A.1.2 note that the effect of this rod must be considered as well as the worth of the most reactive rod that is neglected as a safety analysis assumption.
12-04	A	Consistent with NUREG-1431, rods must be OPERABLE and properly positioned. A misaligned rod is not necessarily inoperable. Modifying the ACTION for a misaligned rod to remove the notion that it is also, by definition, inoperable is consistent with the ITS approach.
12-05		Not Used.
12-06	A	The proposed deletion of the requirement to declare the misaligned rod "inoperable" and the addition of requirements to either verify SDM or initiate boration for a misaligned [or inoperable] rod are consistent with NUREG-1431. In practice, if compliance with the SDM were not verified, boration would be immediately initiated. Because the time typically required to recover the SDM is short, the SDM would typically be determined to be within its limit within the required time frame. Therefore, the additional note to either verify SDM or initiate boration is an administrative change.
12-07	A	Table 3.1-1, "Accident Analyses Reevaluation in the Event of an Inoperable [Full-Length] Rod," would be deleted and replaced with a more general requirement to evaluate accidents to assure the results remain valid. This is acceptable because the general requirement is sufficient to assure that the affected accident analyses will be considered and the ITS Bases discuss the accident analyses affected by rod misalignment.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
12-08	LS9	Consistent with NUREG-1431, the requirement to reduce the high neutron flux set point to ≤ 85 percent of RTP would be deleted. This is acceptable because the underlying safety limits are not of a nature that requires immediate shutdown of the plant if they are exceeded. This is evidenced by the allowance of 72 hours to verify peaking factors. It is assumed that during this 72-hour period an event will not occur which will raise the power level and cause a high neutron flux trip at 100 percent RTP. If a power excursion would occur from the 75 percent RTP ACTION statement limit, the initial peaking factors would not be critical to the analysis, since the analysis is based on the peaking factors at 100 percent RTP. Therefore, the risk of a reactor trip caused by adjusting the power range trip set points is not justified by the potential consequences of failing to reduce the trip set points.
12-09	M	Not applicable to DCCP. See Conversion Comparison Table. (Enclosure 3B)
12-10	LS10	The requirement to maintain RCS $T_{avg} \geq 541^{\circ}\text{F}$ during rod drop testing would be revised to maintain $T_{avg} \geq 500^{\circ}\text{F}$. NUREG-1431, allows the tests to be performed at temperatures as low as 500°F . Because the RCS coolant is more dense at lower temperatures, the rod drop time would be greater at the lower temperatures than at the higher temperatures. In addition, the RCS is borated such that the SDM remains within its limits at the Conditions existing during these tests. Nevertheless, this change, which allows more flexibility of plant conditions for conducting rod drop testing, is a relaxation in plant operations in the CTS.
12-11	TR3	It is proposed to move the requirement to perform rod drop testing on individual rods following maintenance that could affect the drop time to licensee controlled documents [and to delete the 18 month requirement]. The requirement to perform drop time testing following each removal of the reactor vessel head would not be modified. The proposed change is justified, because in addition to being consistent with NUREG-1431, Rev. 1, good maintenance practices would require a retest following any maintenance on a rod or its drive system that could affect drop time. Furthermore, it is difficult to postulate any maintenance on a rod that could affect its drop time without requiring the reactor vessel head to be removed in the process. The components of the rod and rod drive system that affect drop time (as defined in TS) are all inside the reactor coolant pressure boundary. Therefore, moving this requirement outside the TS would have essentially no impact on rod OPERABILITY. [Measuring rod drop time following each removal of the reactor vessel head is considered equivalent if not more restrictive than an 18 month frequency requirement; therefore, deleting the 18 month requirement where it exists (not all plants have it in the CTS) is an administrative change.]

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
12-12	LS13	CTS [3.1.3.1] ACTIONS are revised to delete reference to causes of control rod inoperability due to rod control urgent failure or other electrical problems in the rod control system.
12-13		Not used.
12-14	M	This wording is broadened from "untrippable" to "inoperable" to ensure all causes of inoperability are covered. The previous wording covered inoperable rods if they were untrippable (e.g., "immovable as a result of excessive friction or mechanical interference...") but did not cover trippable rods with drop times that exceed the surveillance limit. These slow rods are inoperable. This more restrictive change clarifies the appropriate ACTIONS to be taken for all causes of inoperability, consistent with Traveler TSTF-107.
12-15	A	Not applicable to DCPD. See Comparison Table (Enclosure 3B).
12-16	LG	Several surveillances (e.g., rod position deviation monitor and rod insertion limit monitor in this section) contain ACTIONS in the form of increased surveillance frequency to be performed in the event of inoperable alarms. These ACTIONS are moved from the CTS to license controlled documents since the alarm does not, in itself, directly relate to the limits. This detail is not required to be in the TS. Therefore, moving this detail is acceptable and is consistent with Traveler TSTF-110, Rev. 1.
12-17	A	Editorial changes are made for clarity. Untrippable rods are addressed through ACTION A; hence, there is no additional need to exclude those rods from these required ACTIONS.
12-18	LG	The technical contents of the ACTION statement which allow continued power operation with a misaligned rod are incorporated into LCO 3.1.4, ACTION B.1 Bases, of the ITS.
12-19	LS18	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
12-20	A	The ACTION statement in the CTS to restore the rod drop time to within limits as a Condition for MODE 2 is captured in the frequency for the performance of ITS SR 3.1.4.3.
12-21		Not used.
12-22	M	This change, in accordance with NUREG-1431, provides a new ACTION in the event the allowed outage times are not met for the rod misalignment action. Prior to this change, LCO 3.0.3 would have been entered allowing for 1 hour prior to placing the plant in HOT STANDBY within the next 6 hours. This change is more restrictive in that the 1 hour time frame is eliminated.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1

(Continued)

**CHANGE
NUMBER**

NSHC

DESCRIPTION

13-01	LG	Consistent with NUREG-1431, the OPERABILITY attributes of equipment and components are described in the Bases. The proposed elimination of the accuracy attributes of the DRPI system and demand position indication system from the specification on position indicating systems would have no impact on OPERABILITY of these systems because the design of these systems is fixed. Furthermore, ITS LCO 3.1.4 requires that all individual indicated rod positions be within 12 steps of their group step counter demand position. Therefore, the LCO effectively establishes the accuracy requirements for the rod position indicating system, and eliminating it from the specification for indicating systems would have no effect on the OPERABILITY of the indicating systems.
13-02	LS15	The requirement for inoperable DRPI is changed from "with a maximum of one per bank" to "one per group for one or more groups." This change is consistent with NUREG-1431. However, the change from "bank" to "group" could potentially result in additional rods with inoperable DRPI because the plant has more groups of rods than banks. Entry into this condition for more than one rod per bank is acceptable because the required ACTIONS, if applied to each non indicating rod, provide appropriate compensatory actions. Position verification of the nonindicating rod within each group within 8 hours is adequate to allow continued full power operation since this frequent monitoring ensures that the probability of having a rod significantly out of position and the simultaneous occurrence of an event sensitive to rod position is small.
13-03	LS12	Consistent with NUREG-1431, a 4 hour completion time is specified to verify rod position after movement of a rod with inoperable indicators more than 24 steps in one direction. This is justified in the ITS Bases as an acceptable Completion Time to perform the verification of rod position using the movable incore detectors. The CTS requirement to perform the verification "immediately" is intended to reflect a quick starting time. A completion time of 4 hours also would assure a prompt start time as stated in the ITS Bases; however, since a flux map can be obtained in less than 4 hours, the change could represent a relaxation in start time.
13-04	M	The ITS specify that the plant be brought to MODE 3 within 6 hours if the Required ACTION and Completion Times are not met. The current specifications do not specify an ACTION if the ACTIONS are not met within the required completion time. The ITS Bases state that 6 hours is a reasonable time, based on operating experience, for reaching MODE 3 from full power conditions without challenging plant systems. The CTS would require entry into TS 3.0.3 which would require achieving MODE 3 within 7 hours. Therefore, the plant shutdown to MODE 3 requirement would be required in 1 less hour.
13-05	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
13-06	A	Not applicable to DCPP. See Conversion Comparison Table. (Enclosure 3B)
13-07	M	The proposed modifications to the SR would require a verification of agreement between digital and demand indicator systems prior to criticality after each removal of the reactor vessel head, instead of every 12 hours. This reflects a reorganization of SRs in the ITS. The requirement for a 12 hour comparison would be moved to SR 3.1.4.1 in the ITS. The post-vessel head removal requirement would be a new specification that demonstrates rod position system OPERABILITY based on a comparison of indicating systems throughout the full range of rod travel. The Frequency requirement of prior to criticality after each removal of the reactor vessel head would permit this comparison to be performed only during plant outages that involve plant evolutions (vessel head removal) that could affect the OPERABILITY of the rod position indication systems. The Frequency change is based on Traveler TSTF-89.
13-08	LS20	Adds provision from Callaway's current specifications which would, under certain conditions, allow continued operation with more than one inoperable DRPI per group. A separate Condition entry allowance is permitted for each inoperable rod position indicator per group and each demand position indicator per bank. A proposed Traveler Westinghouse Owners Group (WOG) - 73, Rev. 1 is in processing to cover this issue.
13-09	LS23	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
14-01	R	The Shutdown Position Indication System Specification 3.1.3.3 is relocated outside of the TS. This is consistent with NUREG-1431.
15-01	R	The Rod Drop Time Specification 3.1.3.4 is relocated outside of the TS. The RCS temperature limit and reactor coolant pumps operating requirement for rod drop testing are combined with CTS Surveillance 4.1.3.4, then incorporated into ITS SR 3.1.4.3. This is consistent with NUREG-1431.
15-02	A	The Rod Drop Time SR 4.1.3.4.a is moved to the Control Rod ITS LCO 3.1.4 as SR 3.1.4.3. This change is consistent with NUREG-1431.
16-01	LS14	This TS would be revised to apply to shutdown "banks" instead of shutdown "rods;" this is consistent with NUREG-1431. The current ACTION statement permits one rod to be inserted beyond the limits; the proposed ITS Condition A would allow one or more banks to be inserted beyond the limit.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
16-02	M	The proposed changes to the ACTION statement would require that the shutdown banks be aligned within limits and that SDM be verified or restored. The new ACTION statement would extend the time to achieve alignment from 1 to 2 hours as justified in the Bases for ITS 3.1.5. The new ACTION statement would establish a Completion Time of 1 hour for verifying and restoring SDM. In the proposed ACTION statement, both the realignment and the SDM verification would be required. The current ACTION statement provides a 1 hour limit to achieve realignment and effectively applies a 2 hour Completion Time to SDM verification and restoration (which would be performed under the TS for rod group alignment limits). In the CTS, either the realignment or the SDM verification are required. The CTS could, in some circumstance, allow continued POWER OPERATION with a shutdown rod out of alignment because it was written to apply to individual rods and refers to the rod group alignment specification. The new ACTION statement, which applies to shutdown banks, would not permit operation with a shutdown bank outside its insertion limits. Overall, the proposed requirements are more restrictive.
16-03	LS22	The CTS SR requiring shutdown rod insertion limits to be verified within 15 minutes prior to withdrawing control rods during approaches to criticality would be deleted. This requirement essentially has been incorporated into LCO 3.1.5 Applicability, which would require that the shutdown banks meet the LCO when in MODE 2 with any control bank not fully inserted. The requirement to perform this check prior to entering the LCO Applicability is covered in plant procedures for startups which require the operator to closely monitor rod position.
16-04	M	Consistent with NUREG-1431, the Applicability would be modified to include MODE 2 with any control bank not fully inserted rather than MODE 2 with $k_{eff} \geq 1.0$. This is more restrictive than the current Applicability.
16-05	M	This change, in accordance with NUREG-1431, provides a new ACTION in the event the AOT are not met for the restoration of the shutdown banks to their insertion limit and the verification that the SDM is within its COLR limit or restored to its limit. Prior to this change, LCO 3.0.3 would have been entered allowing 1 hour prior to placing the plant in HOT STANDBY within the next 6 hours. This change is more restrictive in that the 1 hour time frame is eliminated.
16-06	A	Consistent with NUREG-1431, this change eliminates an unnecessary reference to a separate LCO (see also CN 16-02-M). This change adds the ACTIONS contained in another specification. Rather than reference the other specification, the relevant requirements would simply be inserted into this specification. Therefore, there are no technical differences introduced by this change.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
17-01	M	The proposed modification would add "sequence and overlap" as well as insertion limits to the LCO. These changes do not add any new requirements; as sequence, overlap, and insertion limits are currently specified in the COLR figure defining rod insertion limits vs RATED THERMAL POWER for that portion of the LCO Applicability where these limits provide safety analysis inputs (at or above hot zero power, i.e., \geq 0% RTP. [The COLR will be revised to address the entire CTS LCO 3.1.3.6 Applicability]. The proposed modification would include addition of an SR to periodically verify the overlap and sequence limits that were specifically added per ITS.
17-02	M	This proposed change adds a requirement, that if the bank insertion limits are exceeded, to verify that the SDM is within limits, or initiate boration to restore the SDM to within limits.
17-03	LS21	This change adds an additional 2 hours to the AOT to be in MODE 3. This is consistent with NUREG-1431.
17-04	LS8	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(10 pages)

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.1

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 LG	In accordance with TSTF-9, Rev 1 this change would move the specified limit for SDM from TS to the COLR.	Yes	No, already in CTS.	Yes	Yes
01-02 M	MODE 2 with $K_{eff} < 1.0$ would be added to Applicability of the SDM specification.	No, different CTS	No, different CTS.	Yes	Yes
01-03 LS1	The ACTION statement would be modified to reflect that the requirement to initiate boration at a specified rate with fluid at a specified boron concentration is generalized to simply require boration.	Yes	Yes	Yes	Yes
01-04 LS2	The requirement of SR to verify SDM within 1 hour of detecting an inoperable rod and once per 12 hours thereafter would be deleted from the SDM LCO.	Yes	Yes	Yes	Yes
01-05 LG	The list of specific items to be considered in the performance of a SDM verification is moved to the ITS 3.1.1 Bases.	Yes	Yes	Yes	Yes
01-06 A	This change revises the SDM LCO Applicability to MODE 2 with $K_{eff} < 1.0$, MODE 3, and MODE 4. This change also creates a new core reactivity LCO based on ITS 3.1.2.	Yes	Yes	No, see CN 1-02-M and CTS.	No, see CN 1-02-M and CTS.
01-07 LS16	Changes the time required for the initiation of boration from "immediately" to "within 15 minutes."	Yes	Yes	No, already in CTS.	No, already in CTS.
01-08 A	The technical contents of this SR (verification of SDM through compliance with rod insertion limits) in MODE 1 and MODE 2 with $K_{eff} \geq 1.0$ have been incorporated into LCO 3.1.6 of the ITS.	Yes	Yes	No, see Amendment 89.	No, see Amendment 103.
01-09 A	Moves requirement to verify estimated critical rod position prior to criticality to ITS SR 3.1.6.1.	Yes	Yes	No, see Amendment 89.	No, see Amendment 103.
01-10 M	CTS SR 4.1.1.1.e is modified by this change to be applicable to MODE 2 with $k_{eff} < 1.0$ as well as the current MODES 3 and 4. This is consistent with NUREG 1431. See also CN 01-06-A.	Yes	Yes	No, see Amendment 89.	No, see Amendment 103.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-01 A	This LCO will be combined with the SDM - $T_{avg} > 200^{\circ}\text{F}$ LCO, in accordance with Traveler TSTF-136.	Yes	Yes	Yes	Yes
03-01 A	The footnote referring to CTS special test exceptions would be deleted.	Yes	Yes	Yes	Yes
03-02 LS3	ACTION Statement A.1 of TS for MTC would be revised to require achieving MODE 2 with $k_{eff} < 1.0$ instead of achieving HOT STANDBY if the BOL MTC limit is exceeded and revised rod withdrawal limits cannot be established.	Yes	Yes	Yes	Yes
03-03 A	The statement that administrative withdrawal limits required to meet ACTION Statement A.1 are in addition to insertion limits of another specification would be removed. This is an example of a redundant requirement to meet another TS.	Yes	Yes	Yes	Yes
03-04 LG	ACTION Statement A.2 is addressed prior to exiting ACTION Statement A.1, as discussed in the ITS 3.1.3 Bases.	Yes	Yes	Yes	Yes
03-05 TR2	The requirement to submit a special report to the NRC would be deleted.	Yes	Yes	Yes	Yes
03-06 LS4	This change would incorporate the provisions of ITS 3.1.3 for suspension of MTC testing near the end of the cycle when further significant changes to the MTC would not occur and result in exceeding the EOL limit.	Yes	Yes	Yes	Yes
03-07 LG	The negative EOL moderator temperature limit is moved to the COLR.	Yes	No, already moved to the COLR.	No, already moved to the COLR.	No, already moved to the COLR.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
04-01 LS5	This change would alter the ACTION statement shutdown requirement time limit from a combination of 15 minutes to restore T_{avg} to within limits followed by 15 minutes to be in MODE 3, if T_{avg} could not be restored, to a single 30 minute limit to exit the Applicability if T_{avg} were not within its limit. In addition, the ACTION statement would be changed to require achieving MODE 2 with $k_{eff} < 1.0$ instead of achieving HOT STANDBY if the LCO is not met (refer to TSTF-26).	Yes	Yes	Yes	Yes
04-02 LS6	The SR interval to measure RCS loop average temperature is revised to 12 hours in accordance with industry Traveler TSTF-27, Rev 2.	Yes	Yes	Yes	Yes
04-03 A	The LCO for Minimum Temperature for Criticality, CTS 3.1.1.4, is moved to ITS 3.4.2 in the RCS Section.	Yes	Yes	Yes	Yes
05-01 M	The SR that requires a comparison of measured reactivity to predicted would be modified to add the requirement to compare core reactivity against the predicted prior to entry into MODE 1 after refueling outages.	Yes	Yes	Yes	Yes
05-02 LS7	This proposed change would specify that the overall core reactivity balance comparison shall be done every 31 EFPD after burnup exceeds 60 EFPD.	Yes	Yes	Yes	Yes
05-03 LG	The list of specific items to be considered in the performance of a reactivity balance verification is moved to the ITS 3.1.2 Bases.	Yes	Yes	Yes	Yes
05-04 A	The SR requiring SDM to be verified prior to initial operation in MODE 1 after each refueling is effectively performed under other specifications. This SR required a SDM verification prior to operation above 5 percent power after each refueling with the control rod banks at maximum insertion limits and is being deleted since it is redundant.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.1

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-05 LS17	ACTIONS to be taken should the reactivity balance not be within limits are provided, in lieu of a TS 3.0.3 shutdown.	Yes	Yes	No, already in CTS.	No, already in CTS.
05-06 A	CTS SR 4.1.1.5.1 requires that the predicted reactivity values "shall" be adjusted (normalized) at 60 EFPD after refueling. ITS SR 3.1.2.1 states the normalization requirement as "may" be adjusted: This is to recognize that normalization is not necessary if predicted and measured core reactivity are within acceptance tolerance. The scheduling of predicted and measured core reactivity continues to be required at 60 EFPD. Therefore, this change reflects clarification of existing intent and is considered administrative.	No, maintaining CTS wording.	No, maintaining CTS wording.	Yes	Yes
06-01 R	Relocates "Boration Flow Path - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 1.	Yes	No, see Amendment 89.	No, see Amendment 103.
07-01 R	Relocates "Boration Flow Path - Operating" TS to licensee controlled document.	Yes, see LAR 95 07 dated 10/4/95, DCL 95-222.	Yes	No, see Amendment 89.	No, see Amendment 103.
07-02 A	Moves limitation on charging pumps in MODE 4 to ITS SR 3.4.12.2.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-01 R	Relocates "Charging Pumps - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 3.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-02 M	Moves charging pump SR when below 350°F to ITS SR 3.4.12.2 and decreases surveillance frequency to 12 hours from 31 days.	No, already in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.
08-03 LS19	Deletes the method of verificating that charging pumps are incapable of injecting into the RCS.	No, not in CTS.	Yes	No, see Amendment 89.	No, see Amendment 103.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-01 R	Relocates "Charging Pumps - Operating" TS to licensee controlled document.	Yes, see Attachment 21, Page 5.	Yes	No, see Amendment 89.	No, see Amendment 103.
10-01 R	Relocates "Borated Water Source - Shutdown" TS to licensee controlled document.	Yes, see Attachment 21, Page 7.	Yes	No, see Amendment 89.	No, see Amendment 103.
11-01 R	Relocates "Borated Water Source - Operating" TS to licensee controlled document.	Yes, see Attachment 21, Page 9.	Yes	No, see Amendment 89.	No, see Amendment 103.
12-01 A	The reference to "full-length" rods would be deleted.	Yes	No, not in CTS.	Yes	Yes
12-02 M	The ACTIONS required for more than one misaligned, but operable, rod would be changed to be identical to those for inoperable rods.	Yes	Yes	Yes	Yes
12-03 A	The requirement to include an increased allowance in the SDM calculation for the untrippable control rod is inherent in the SDM Definition.	Yes	Yes	Yes	Yes
12-04 A	ACTION for a misaligned rod would be modified to eliminate the notion that a misaligned rod is, by definition, inoperable.	Yes	Yes	Yes	Yes
12-05	Not used.	N/A	N/A	N/A	N/A
12-06 A	ACTION for a misaligned rod would be modified to require boration to restore SDM if not within limits.	Yes	Yes	Yes	Yes
12-07 A	Table 3.1-1, "Accident Analyses Requiring Reevaluation in the Event of an "Inoperable [Full-Length] Rod" would be eliminated.	Yes	Yes	Yes	Yes
12-08 LS9	The requirement to reduce the high neutron flux set point to \leq 85% of RTP would be deleted.	Yes	Yes	Yes	Yes

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-09 M	This proposed change would reinstate an SDM verification requirement that had been eliminated in a previous license amendment.	No, already in CTS.	No, already in CTS.	Yes	Yes
12-10 LS10	The requirement to maintain $RCS T_{avg} \geq [541]^{\circ}F$ during rod drop testing would be revised to maintain $T_{avg} \geq 500^{\circ}F$.	Yes	Yes	Yes	Yes
12-11 TR3	The requirement to perform drop testing on rods following maintenance would be removed from the CTS.	Yes, also deletes redundant 18 month interval.	Yes, also deletes redundant 18 month interval.	Yes	Yes
12-12 LS13	CTS [3.1.3.1] ACTIONS are revised to delete reference causes of control rod inoperability due to rod control urgent failure or other electrical problems in the rod control system.	Yes	Yes	Yes	No, CTS already revised to incorporate.
12-13	Not used.	N/A	N/A	N/A	N/A
12-14 M	This wording is broadened from "untrippable" to "inoperable" to ensure all causes of inoperability are covered. This more restrictive change clarifies the appropriate ACTIONS to be taken for all causes of inoperability, consistent with Traveler TSTF-107.	Yes	Yes	Yes	Yes
12-15 A	Rod misalignment is determined based on a comparison between the rod's DRPI and its group step counter demand position, not on a rod to rod position verification. This change is administrative in nature in that there is no effect on the manner in which the operating staff would determine whether a misalignment event had occurred.	No, already in CTS.	No, already in CTS.	Yes	Yes
12-16 LG	Several surveillances (e.g., rod position deviation monitor and rod insertion limit monitor in this section) contain ACTIONS in the form of increased surveillance frequency to be performed in the event of inoperable alarms. These ACTIONS are relocated from the TS to licensee controlled documents. This is consistent with TSTF-110, Rev. 1.	Yes, moved to the FSAR.	Yes, moved to TRM.	Yes, moved to the USAR.	Yes, moved to the FSAR, Section 16.1.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-17 A	Editorial changes are made for clarity. Untrippable rods are addressed through ACTION A; hence, there is no additional need to exclude those rods from these Required Actions.	Yes	Yes	No, not in CTS.	No, not in CTS.
12-18 LG	The technical contents of the ACTION statement which allow continued POWER OPERATION with a misaligned rod are moved to the Bases for ITS LCO 3.1.4, ACTION B.1.	Yes	Yes	Yes	Yes
12-19 LS18	The frequency at which the rod motion surveillance is performed is extended from 31 days to 92 days.	No, already in CTS.	Yes	Yes	No, already in CTS.
12-20 A	The ACTION Statement in the CTS to restore the rod drop time to within limits as a Condition for MODE 2 is captured in the frequency for the performance of ITS SR 3.1.4.3.	Yes	Yes	No, see Amendment 89.	No, see Amendment 103.
12-21	Not used.	N/A	N/A	N/A	N/A
12-22 M	This change, in accordance with NUREG-1431, provides a new ACTION in the event the AOTs are not met for the rod misalignment action. Prior to this change, LCO 3.0.3 would have been entered allowing for 1 hour prior to placing the plant in HOT STANDBY within the next 6 hours. This change is more restrictive in that the 1 hour time frame is eliminated.	Yes	Yes	Yes	Yes
13-01 LG	The specific OPERABILITY attributes of the DRPI system would be moved to the Bases.	Yes	Yes	Yes	Yes
13-02 LS15	The requirement for inoperable DPRI is changed from "with a maximum of one per bank" to "one per group for one or more groups."	Yes	Yes	Yes	Yes
13-03 LS12	A 4 hour Completion Time is specified to verify rod position after movement of a rod with inoperable indicators more than 24 steps in one direction.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.1

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
13-04 M	A requirement would be added to bring the plant to MODE 3 within 6 hours if the required ACTIONS and Completion Times are not met.	Yes	Yes	Yes	Yes
13-05 A	The proposed change would retain an ACTION statement, currently in the plant TS, that permits continued POWER OPERATION with more than 1 digital rod position indicator per group inoperable.	No, not in CTS - see 13-08-LS20.	No, not in CTS - see 13-08-LS20.	Yes	Yes
13-06 A	The change would allow separate Condition entry for each inoperable DRPI per group or each demand indicator per bank.	No, not in CTS - see 13-08-LS20.	No, not in CTS - see 13-08-LS20.	Yes	Yes
13-07 M	The proposed modifications to the SR would verify agreement between digital and demand indicator systems prior to criticality after the reactor vessel head was removed instead of every 12 hours. The Frequency change is based on Traveler TSTF-89.	Yes	Yes	Yes	Yes
13-08 LS20	Adds provision in Callaway's current specifications which would, under certain Conditions, allow continued operation with more than one inoperable DRPI per group. This is consistent with Traveler WOG-73, Rev. 1.	Yes	Yes	No, already in CTS.	No, already in CTS.
13-09 LS23	CTS ACTIONS b.1.b) and b.1.c) of LCO 3.1.3.2 are deleted. SDM is ensured in MODES 1 and 2 by rod position. Multiple inoperable DRPIs will have no impact on SDM in MODES 1 and 2 if the control rod positions are verified by alternate means and rod motion is limited consistent with the accident analyses. Deletion of these requirements is consistent with traveler WOG-73, Rev. 1.	No, not in CTS.	No, not in CTS.	Yes	Yes
14-01 R	Relocates CTS 3.1.3.3 to licensee controlled documents, consistent with NUREG-1431.	Yes, see LAR 95-07 dated 10/4/95, DCL 95-222.	Yes, relocated to TRM.	No, see Amendment 89.	No, see Amendment 103.

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
15-01 R	The Rod Drop Time Specification 3.1.3.4 is relocated outside of the TS. The RCS temperature limit and RCPs operating requirement for rod drop testing are combined with CTS Surveillance 4.1.3.4 then incorporated into ITS SR 3.1.4.3.	Yes, see LAR 95-07 dated 10/4/95, DCL 95-222.	No, not in CTS - see CN 15-02-A.	No, see Amendment 89.	No, see Amendment 103.
15-02 A	The Rod Drop Time SR 4.1.3.4.a is moved to the Control Rod ITS LCO 3.1.4 as SR 3.1.4.3.	Yes	Yes	No, already in CTS.	No, already in CTS.
16-01 LS14	The requirement for shutdown insertion limits would be applied to shutdown banks rather than shutdown rods.	Yes	Yes	Yes	Yes
16-02 M	ACTION statements would be changed to specify 2 hours to achieve rod alignment and to prohibit POWER OPERATION with a shutdown bank outside insertion limits.	Yes	Yes	Yes	Yes
16-03 LS22	The requirement to verify shutdown bank insertion within 15 minutes prior to withdrawing any control bank rods during startup would be deleted.	Yes	Yes	Yes	Yes
16-04 M	The Applicability would be modified to include MODE 2 with any control bank not fully inserted.	Yes	Yes	Yes	Yes
16-05 M	This change provides a new ACTION in the event the AOTS are not met for the restoration of the shutdown banks to their insertion limit. Prior to this change, LCO 3.0.3 would have been entered allowing 1 hour prior to placing the plant in HOT STANDBY within the next 6 hours. This change is more restrictive in that the 1 hour time frame is eliminated.	Yes	Yes	Yes	Yes
16-06 A	This change eliminates an unnecessary reference to a separate LCO.	Yes	Yes	Yes	Yes

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
17-01 M	Requirements for sequence and overlap, as well as insertion limits, would be included in the control rod insertion TS. An SR to check the sequence and overlap, as well as insertion limits, would be included in the TS also.	Yes	Yes	Yes	Yes
17-02 M	Adds a requirement that, if the bank insertion limits are exceeded to verify that the SDM is within limits, or initiate boration to restore the SDM to within limits.	Yes	Yes	No, already in CTS.	No, already in CTS.
17-03 LS21	This change adds an additional 2 hours to the AOT to be in MODE 3.	Yes	Yes	Yes	Yes
17-04 LS8	The Wolf Creek CTS ACTION [3.1.3.6.d] is revised to be "Be in MODE 2 with $k_{eff} < 1.0$." consistent with ITS 3.1.6 to place the unit in a MODE in which the LCO does not apply.	No	No	Yes	No

ENCLOSURE 4
NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3..." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R" (Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1

10CFR50.92 EVALUATION
FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The ACTION statement would be modified to reflect that the requirement to initiate boration at a specified rate with fluid at a specified boron concentration is generalized to simply require boration. As described in the ITS Bases, the required flow rate and boron concentration should be selected depending on plant conditions, available equipment, and the magnitude of the deviation between the actual and required SDM. The ITS Bases allow the operator to use the "best source available for the plant conditions." This is an example of maintaining the overall safety requirement in TS but removing procedural details from the TS allowing the plant operator the ability to select the appropriate procedure and equipment for the existing plant condition.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The ACTION statement that would be modified requires that SDM be restored if not within limits by boration at a specific flow rate with a specific concentration of boron. The proposed change would eliminate details regarding flow rates and concentrations from the TS; however the overall requirement to restore SDM would remain. Removal of the details of system operation would not affect the probability of an accident occurring. For consequences of an accident to be affected, an accident that is impacted by SDM would have to occur during the short 1 or 2 hour time period that SDM is not within limit. The probability of that occurrence is negligible because of the short duration expected to return SDM to within limits. Therefore, there would be no significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes may affect the timing of SDM restoration depending on the equipment selected by the operator to perform the boration. The change involves no hardware modifications or changes in the manner in which plant systems perform their functions. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

There will be no reduction in any margin of safety. The SDM limits are not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_Q , F_{2H}^N , LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS1" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A surveillance in the SDM specification requires SDM to be verified in the event of detecting an inoperable rod. Since it applies to inoperable (untrippable) rods, it should only be applicable in MODES 1 and 2 []. The rods are used to maintain SDM in MODES 1 and 2 by maintaining the rods within insertion limits. The requirements for rod alignment limits in CTS specify ACTIONS to be taken upon detecting an inoperable rod. The ACTIONS include verifying SDM within 1 hour. Deletion of the CTS surveillances to verify SDM when a rod(s) is inoperable in the shutdown MODES [3 through 5] accomplishes the following:

- 1) Deletes an inappropriate Applicability statement. The surveillance should not apply in those MODES when the rods are not required to be OPERABLE and,
- 2) Deletes redundant requirements. The requirements are properly and fully addressed in the specifications related to rod alignment OPERABILITY and insertion limits.

If a rod is declared inoperable after shutting down to MODE 3 (or other shutdown modes), the TS definition of SDM requires the inoperable rod to be accounted for in the SDM calculation per ITS SR 3.1.1.1. If a rod were to become inoperable during rod testing in the shutdown modes, there would be no adverse effects on safety because the definition of SDM accounts for a rod failure prior to performing the tests.

Implicit in ITS LCO 3.1.1 and SR 3.1.1.1 is the need to reperform the SDM calculation if any of these effects have changed since the last 24 hour calculation.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Elimination of the SRs would have no effect on the probability of an accident. The requirements for maintaining SDM with an untrippable rod and during rod testing assure that there would be no effect on the consequences of any accidents for which SDM is a factor. Therefore, there would be no increase in the probability or consequences of a previously evaluated accident as a result of eliminating the SR.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The SR being eliminated by this change requires periodic checks of SDM. Eliminating these checks would not affect the manner of operation of the plant or any plant systems so that the occurrence of a new kind of accident could result. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

As noted in the introductory paragraphs above, the TS definition of SDM requires the inoperable rod to be accounted for in SDM calculations. Further, SDM accounts for rod movement during testing, so an untrippable rod during testing would not have an adverse impact on SDM. As long as the SDM limits continue to be met as required, no reduction in a margin of safety could result. Based on the previous discussion, the plant conditions required for rod testing and by other specifications assure the SDM requirements would be met. The SDM limits are not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_{Q, F_{dH}^N} , LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS2" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

ACTION Statement A.1 would be revised to require achieving MODE 2 with $k_{\text{eff}} < 1.0$ instead of achieving HOT STANDBY if the BOL MTC limit is exceeded and revised rod withdrawal limits cannot be established within 24 hours. This change corrects the discrepancy between the BOL Applicability and the ACTION, while ensuring that the plant is taken to a condition in which the LCO is not applicable. Revising the CTS, albeit to correct an inconsistency, represents a relaxation in ACTION Statement A.1.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The ACTION of taking the reactor to MODE 2 with $k_{\text{eff}} < 1.0$ instead of MODE 3 would have no effect on the probability of any accident occurring. MTC is an initial Condition assumed in accident analyses, and therefore if an accident were to occur while the MTC was outside the limits, the consequences of an accident could be exacerbated. However, placing the reactor in MODE 2 with $k_{\text{eff}} < 1.0$ assures that the consequences would be bounded by safety analyses should an accident occur. Therefore, there would be no increase in the probability or consequences of a previously evaluated accident as a result of making the proposed change.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change would not create the possibility of a new or different accident, because the plant or its systems would not be operated differently. The original requirement was to achieve MODE 3, HOT STANDBY. To achieve MODE 3, it is necessary to pass through MODE 2 with $k_{\text{eff}} < 1.0$. Therefore, there are no physical alterations to any plant equipment, and cause no changes in the methods by which any safety system performs its function. There are no changes to the operation of the plant or equipment. Therefore, this change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The margins of safety in question are those in the design basis accident analyses (DBA) that rely on MTC as an initial Condition. As noted in the evaluation of Criterion 1 above, placing the plant in MODE 2 with $k_{eff} < 1.0$ assures that the accidents will not violate the assumptions of the analyses. The MTC limits are not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_Q, F_{DH}^N , LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS3" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change would incorporate the provisions of ITS 3.1.3 allowing suspension of MTC testing near the end of the cycle when further significant changes to the MTC would not occur and result in exceeding the EOL limit. This represents a relaxation in performing the SR.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The change would allow the suspension of surveillance testing for the lower bound MTC limit near the end of reactor core life. In order to implement this change, additional analysis must be performed to define a surveillance MTC value below which the actual MTC would not fall as the core neared its end-of-cycle (EOC) condition. The MTC value, which becomes a limit established in the COLR, is conservatively selected to assure, that if met the MTC would not exceed the EOC limiting value. It is possible to establish a surveillance limit because of the gradual manner in which MTC changes near the end of core life. MTC is an initial condition assumed in accident analyses. Changes in the manner in which MTC is monitored would not affect the probability of an accident. Also, the consequences of accidents that depend on MTC as an initial Condition would not be increased if MTC were maintained within limit. The suspension of surveillance for MTC near the end of core life would be based on a conservative analysis of MTC behavior and operating experience regarding the slow manner in which MTC varies over time near the end of core life. The suspension of surveillance would occur at a time when there is negligible chance that the lower MTC limit value would be exceeded. Therefore, the proposed change in MTC surveillance would involve no increase in either the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes deal with the frequency of monitoring a parameter (MTC) that is an initial condition of accident analyses. The limits on MTC would not be changed. Changes in SR frequency would not lead to changes in plant or plant system operations or other conditions that could cause an accident of a new or different type. Thus, the proposed change does not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

As noted in the evaluation of Criterion 1 above, the proposed changes would not include any changes in the limits for MTC; therefore, the margins of safety afforded by MTC limits would not be affected. The surveillance limit used to define the suspension of MTC monitoring would be selected using conservative analyses and knowledge of the manner in which MTC varies over core life. Therefore, there is little chance that core MTC values could exceed the EOC limit. The MTC limits are not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_{α} , $F_{\Delta H}^N$, LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed change associated with NSHC "LS4" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed change would make two changes to the ACTION statement. First, it would alter the ACTION statement shutdown requirement time limit from a combination of 15 minutes to restore T_{avg} to within limits followed by 15 minutes to be in MODE 3, if T_{avg} could not be restored, to a single 30 minute limit to be in MODE 3 if T_{avg} were not within its limit. Second, the ACTION statement would be revised to require achieving MODE 2 with $k_{eff} < 1.0$ instead of achieving HOT STANDBY if the LCO is not met (refer to TSTF-26). Regarding the first change, both the current requirement and the ITS requirement are essentially equivalent in that the plant is now required to be in MODE 3 within 30 minutes after discovering that a parameter is not within its limits, if the parameter is not restored to within its limits in that 30 minute time period. If the LCO can be satisfied at any time during the 30 minute time frame, the plant shutdown can be terminated. Regarding the second change, it represents a relaxation in current ACTION statement requirements for plant shutdown and is the subject of this evaluation.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The ACTION of taking the reactor to MODE 2 with $k_{eff} < 1.0$ instead of MODE 3 would have no effect on the probability of any accident occurring. T_{avg} is an initial condition assumed in accident analyses, and therefore if an accident were to occur while the T_{avg} was outside the limits, the consequences of an accident could be exacerbated. However, placing the reactor in MODE 2 with $k_{eff} < 1.0$ assures that the consequences would be bounded by safety analyses should an accident occur. Therefore, there would be no increase in the probability or consequences of a previously evaluated accident as a result of making the proposed change.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change would not create the possibility of a new or different accident because the plant or its systems would not be operated any differently. The original requirement was to achieve MODE 3, HOT STANDBY. To achieve MODE 3, it is necessary to pass through MODE 2 with $k_{eff} < 1.0$. Therefore, there would be no operational changes to contribute to the possibility of a new accident resulting from the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The margins of safety in question are those in the DBA analyses that rely on T_{avg} as an initial Condition. As noted in the evaluation of Criterion 1 above, placing the plant in MODE 2 with $K_{eff} < 1.0$ assures that the accidents will not violate the assumptions of the analyses. The minimum T_{avg} for criticality is not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_Q, F_{dH}^N , LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS5" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would revise the SR for verifying that RCS temperature (T_{avg}) is within limits by changing the frequency to once per 12 hours in accordance with TSTF-27, Rev. 2. The current frequency requirements were within 15 minutes prior to achieving reactor criticality, which is redundant and unnecessary since T_{avg} must be within its limit prior to entering the LCO Applicability, and at least once per 30 minutes when the reactor is critical and the ($T_{avg}-T_{ref}$) deviation alarm is not reset. The RCS temperature is maintained within limit: (1) to assure that the MTC is within the limits assumed in the accident analyses, (2) to assure that the neutron detectors are not adversely affected by neutron attenuation caused by low coolant temperature, (3) to assure that the RCS and pressurizer response to thermal-hydraulic transients is as predicted, and (4) to assure that the reactor vessel temperature is above the nil-ductility transition reference temperature.

The plant design incorporates monitoring of T_{avg} and provides an alarm, the ($T_{avg}-T_{ref}$) deviation alarm, as T_{avg} approaches its limit. This alarm Condition requires a response by the operating staff. Therefore, at any time that T_{avg} is approaching its limiting value, the plant operators would receive an alarm and initiate corrective ACTION.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The CTS requires that the RCS average temperature is required to be verified within 15 minutes prior to reactor criticality. No additional verification is required during reactor criticality unless the ($T_{avg}-T_{ref}$) deviation alarm actuated. Therefore, the CTS relies on the ($T_{avg}-T_{ref}$) deviation alarm to assure T_{avg} is within limits during reactor operation. The proposed change would introduce a 12 hour requirement to verify T_{avg} independent of the ($T_{avg}-T_{ref}$) deviation alarm. Therefore, the proposed change would provide additional assurance beyond the CTS SR requirement that T_{avg} was within the limits assumed in accident and transient analyses.

With regard to the requirement to verify temperature within 15 minutes of achieving reactor criticality, the ($T_{avg}-T_{ref}$) deviation alarm will still provide warning that RCS temperature is not within limit. In addition, during an approach to criticality, the plant is operated such that rapid or significant temperature changes in the RCS are avoided. Since the specification has no CTS.4.0.4 exception, T_{avg} must still be within limit prior to entering the LCO Applicability, i.e. prior to criticality.

Based on the foregoing discussion, the proposed change would provide additional assurance that RCS T_{avg} would remain within limits assumed in accident analyses during reactor operation. Therefore, the proposed change would not involve a significant increase in the probability or consequences of a previously evaluated accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change deals with the frequency of monitoring a parameter (RCS temperature) that is an initial Condition of accident and transient analyses. Changes in SR frequency would not lead to changes in plant system operations or other conditions that could cause an accident of a new or different type. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

As noted in the evaluation of Criterion 1 above, the main method of monitoring RCS temperature, during normal plant operation and during approach to criticality, is via the $(T_{avg} - T_{ref})$ deviation alarm. This situation would not be altered by the proposed changes. However, the proposed change would add an additional requirement to verify RCS temperature every 12 hours. The minimum T_{avg} for criticality is not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, $F_Q, F_{\Delta H}^N$, LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS6" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed change would specify that the overall core reactivity balance comparison shall be done every 31 EFPD after burnup exceeds 60 EFPD. This is in accordance with NUREG-1431 and clarifies when the 31 EFPD surveillance should start. The CTS SR specifies a frequency of 31 EFPD. The delay in initiating the monthly surveillance is acceptable because of the slow rate of changes in the core due to fuel depletion and the presence of other indicators for prompt determination of an anomaly. As noted in the Bases for ITS 3.1.3, the reactivity balance normalization must be done within the first 60 EFPDs after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change would delay implementing the periodic surveillance until after the initial 60 EFPD after startup. The current specification requires the SR to be performed every 31 effective full power days (EFPD). The proposed change would delay implementing the periodic surveillance until after the initial 60 EFPD after startup. The Bases to ITS 3.1.2 state that the 60 EFPD would allow sufficient time for core parameters to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. A requirement to perform a reactivity balance prior to entering MODE 1 after each refueling would remain in the TS. Adopting the proposed change to the SR would have a negligible effect on the probability or consequences of accidents. The initial reactivity balance that is performed prior to MODE 1 operation provides assurance that the core design is acceptable and initial safe operation would be assured. During the initial 60 EFPD of operation, other parameters provide indication that the core is operating according to design. These indications include the TS requirements for axial flux difference, quadrant power tilt ratio, and hot channel factors. This provides a check of the core Conditions and design calculations and assures that safe operation may proceed until the core conditions reach steady state. Core conditions are used as input assumptions in accident analyses; they do not affect the probability of occurrence of an accident. While they would affect the consequences of postulated accidents, the initial reactivity balance and the availability of other indicators of core performance assure that the impact on accident consequences would be negligible. Therefore, the proposed change to core reactivity surveillance would have an insignificant effect on the probability or consequences of a previously evaluated accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Core reactivity is used as an initial assumption of accident analyses. The proposed change to the surveillance of core reactivity does not involve changes in the operation of plant systems and equipment such that a new accident could result. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

During the initial 60 EFPD, the margins of safety that depend on core reactivity would be assured by the initial reactivity balance performed prior to MODE 1 operation and by the availability of other indicators of core performance that would alert the operators to an anomaly. There are no changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_Q, F_{dH}^N , LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS7" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9 10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, the requirement to reduce the high neutron flux setpoint to \leq 85 percent of RTP would be deleted. This requirement is one of the provisions for continued plant operation, at reduced power, with a misaligned rod. Removing this requirement is acceptable because the underlying safety limits are not of a nature that require immediate shutdown of the plant if they are exceeded. This is evidenced by the allowance of 72 hours to verify peaking factors. It is assumed that during this 72 hour period an event will not occur which will raise the power level and cause a high neutron flux trip at 100 percent RTP. If a power excursion would occur from the 75 percent RTP ACTION statement limit, the initial peaking factors would not be critical to the analysis; since the analysis is based on the peaking factors at 100 percent RTP. Therefore, the risk of a reactor trip caused by adjusting the power range trip setpoints is not justified by the potential consequences of failing to reduce the trip setpoints.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The concern with operation while a rod is misaligned involves potential reduction in SDM and potential core power distributions that are not consistent with accident analysis assumptions. While the probability of an accident would not be affected by the proposed change, the consequences could potentially be affected if the accident analysis assumptions were violated. However, the remaining ACTIONS required for operation in this condition provide compensatory measures that would assure, that should an accident occur, the consequences would be acceptable. These additional ACTIONS are verification of SDM, reduction in reactor power to 75 percent RTP, development of a power distribution map, and a reevaluation of affected safety analyses. The verification of SDM within 1 hour assures that reactor shutdown can be performed in accordance with the accident analyses. The power distribution map, required within 72 hours, assures that longer-term operation would not violate analysis assumptions. The reduction in power, required within 2 hours, assures that local heat generation rates would not cause core design criteria to be exceeded. Based on these additional ACTIONS, the consequences of an accident would not be significantly changed as a result of the proposed change. Therefore, the proposed TS change would not involve any significant increase in the probability or consequences of a previously evaluated accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change would eliminate a requirement to reset the high neutron flux reactor trip setpoint to a lower value. This would not have any affect on the potential accidents that may occur. The setpoint reduction requirement was commensurate with the requirement to reduce reactor power to 75 percent RTP. Operating the plant under these conditions without a reduced trip setpoint would not result in conditions that could cause a new or different type of accident to occur. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The required reduction in reactor power is intended to assure that the margins of safety in the accident analyses are maintained while operating with a misaligned rod. Also, the other Required ACTIONS are intended to assure the assumptions of the safety analyses remain valid. Eliminating the requirement to reset reactor high power trips would have negligible adverse affect on the margins of safety, because the reduction in power tends to increase the available margins and because of the low probability of occurrence of an event that would be terminated by high neutron flux. The rod OPERABILITY and alignment limits are not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_o, F_{DH}^N , LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS9" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10 10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The requirement to maintain RCS $T_{avg} \geq 541^\circ\text{F}$ during rod drop testing would be revised to maintain $T_{avg} \geq 500^\circ\text{F}$. NUREG-1431 allows the tests to be performed at temperatures as low as 500°F . Because the RCS coolant is more dense at lower temperatures, the rod drop time would be greater at the lower temperatures than at the higher temperatures. In addition, the RCS is borated such that the SDM remains within its limits for the conditions existing during these tests. Nevertheless, this change, which allows more flexibility on plant conditions for conducting rod drop testing, is a relaxation in plant operations under the TS.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves the temperature at which rod drop testing is performed. Rod drop time is assumed as an initial condition of accident analyses that depend on reactor scram to mitigate the event. Performance of rod drop time tests would have no effect on the probability of an accident; therefore, altering the temperature for testing has no effect on the probability of an accident. Also, the performance of the test at lower RCS temperature is more conservative from the standpoint of drop time; thus if the rods pass the test at lower temperature, they will be assured of performing as required at the operating reactor temperatures. Therefore, the rods will remain OPERABLE as assumed in accident analyses. Therefore, the proposed change would not result in a significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves a change to the temperature at which rod drop times are measured. This surveillance does not involve any physical alteration to any plant equipment, and causes no change in the method by which any safety-related system performs its function. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

OPERABILITY of the rods would continue to be demonstrated after the proposed change is implemented. Therefore, the rods would continue to perform the mitigating functions assumed in the safety analyses. The rod OPERABILITY and alignment limits are not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, $F_Q, F_{\Delta H}^N$, LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS10" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS12
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, a 4 hour Completion Time is specified to verify rod position after movement of a rod with inoperable indicators more than 24 steps in one direction. This is justified in the ITS Bases as an acceptable Completion Time to perform the verification of rod position using the movable incore detectors. The current requirement to perform the verification "immediately" is intended to reflect a quick starting time. A Completion Time of 4 hours also would assure a prompt start time as stated in the ITS Bases; however, since a flux map can be obtained in much less than 4 hours, the change would represent a relaxation in start time.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves performing flux mapping to indirectly determine the position of rods with inoperable digital position indication. The intent of the ACTION statement that requires a flux map to be obtained is to assure that the rod positions are known after significant rod motion has occurred. The difference between an "immediate" start of a flux map, specified in the current ACTION, and a 4 hour completion, which is proposed, would mean that the rod position in the latter case may not be known for a longer period of time. The period may even approach 2 or 3 hours. However, as noted in the Bases to ITS 3.1.7, the 4 hour period for verifying rod position has been determined to be acceptable on the basis that operation for this time period would not result in undesirable power distributions in the core that could occur from long-term operation with a misaligned rod. It should be noted that the TS require that the rods also have demand position indication available. Demand position indication would provide additional indication of the position of rods, however, at reduced reliability because demand position is based on signals sent to the rods not on feedback from the rod. However, demand position will provide accurate indication unless a rod is stuck in place. The combination of loss of digital position and one or more sticking rods, and an event that depends on rod position occurring during the period of time (4 hour completion) involved is very unlikely. Therefore, based on the foregoing, the proposed change would not result in a significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves the timing of alternative rod position measurements. The change does not involve any physical alternation to any plant equipment, and causes no change in the method by which any safety-related system performs its function. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS12 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The margins of safety for accident analyses that depend on rod position may be affected by misalignment of rods and by power distributions that have not been assumed in accident analyses. However, in the case of the proposed change, the likelihood of the combination of events required to obtain an actual undetected misalignment is extremely unlikely and the Completion Time proposed has been determined to be acceptable from the standpoint of power distribution. There are no changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, $F_Q, F_{\Delta H}^N$, LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS12" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS13
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

CTS [3.1.3.1] ACTIONS are revised to delete reference causes of control rod inoperability due to rod control urgent failure or other electrical problems in the rod control system. The corresponding ITS 3.1.4 Condition B only address a misaligned rod. NUREG-1431, Rev. 1, is based on the position that the OPERABILITY of both shutdown rods and control rods is solely contingent upon the rod being able to perform its safety function. NUREG-1431 defines the rod safety function as trippability, i.e., capable of being tripped and tripping within the require time interval (rod drop time). Routine movement of control rods is not part of the control rod safety function. This change is acceptable since ITS 3.1.4 clarifies rod OPERABILITY, and focuses the ACTIONS on rod untrippability and rod misalignment, consistent with the intended safety functions of the control and shutdown rods.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The proposed change to the TS does not involve a significant hazards consideration based on the evaluation of the proposed operation of the plant in accordance with the following criteria:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not involve any hardware changes or impact the ability of the control rods to perform their safety function. In the event of a rod control urgent failure alarm or other electrical problem in the rod control system, routine control rod movement is impacted. However, routine movement is not part of the safety function of the control rod. Routine movement may be necessary to assure compliance with other reactivity control and power distribution limit specifications. In the event of failure to satisfy one of these other specifications, appropriate compensatory measures are provided in those specifications. The proposed change also does not affect the compensatory measures required for a misaligned control rod. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in controlling parameters. The proposed change will still ensure the control rod safety function is capable of being performed and the proposed change does not affect the actions provided for a misaligned control rod. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

3. Does this change involve a significant reduction in a margin of safety?

The deletion of the condition for control rod inoperability due to rod control urgent failure or other electrical problems in the rod control system is considered acceptable since in this condition the rods are still capable of performing their safety function. Additionally, all aspects associated with the inability to move the control rods are adequately addressed in the ACTIONS of the affected Reactivity Control or Power Distribution Limits TS and the appropriate ACTIONS provided for a misaligned rod. Since the ability of the control rods to perform their safety function remains unchanged, there is no change in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS13" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The specification ACTION statement would be revised to apply to shutdown "banks" instead of shutdown "rods." This change is consistent with NUREG-1431. The CTS ACTION statement permits one or more "rods" to be inserted beyond the limits. The proposed ITS Condition A would allow 1 or more "banks" to be inserted beyond the limit. Permitting a bank to be inserted beyond the limits would likely involve more than 1 rod. Therefore, the proposed specification would be less restrictive than the current one.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The shutdown rods positioned within insertion limits are a major accident mitigation system for accidents that require a reactor trip. However, the shutdown rod position would have no effect on initiating an accident requiring reactor trip. Thus, the proposed changes would have no effect on the probability of accidents previously evaluated. The consequences of accidents requiring reactor trip or a rod ejection accident may be made worse by having a bank of rods inserted beyond the insertion limits. However, the consequences are not significant because the proposed TS changes would incorporate a provision to verify SDM or initiate boration within one hour of detecting the misaligned bank of rods. Thus, for those events which rely on the shutdown rods providing SDM, the required SDM would, within a short time, be restored. The probability that an accident requiring reactor trip would occur during the time that it would take to either restore the bank to within limits or to borate to restore SDM would be extremely small. Therefore, the proposed change would not result in a significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not involve any physical alteration to any plant equipment, and causes no change in the method by which any safety-related system performs its function. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

As discussed in the evaluation of Criterion 1 above, the proposed changes include compensatory measures to be taken if a shutdown bank is discovered to be inserted beyond the limits. The purpose of these measures is to regain the margin in safety analyses that may be lost because of the reduced worth of the shutdown rods. These measures include a time period for their accomplishment, and should the time requirement not be met, the plant must be shut down. Therefore, the total scope of the proposed change includes mitigating factors to assure safety margins are met if continued plant operation is planned. The shutdown rod insertion limits are not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_o, F_{DH}^N , LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS14" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS15
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The requirement for inoperable DRPI is changed from "with a maximum of one per bank" to "one per group for one or more groups." This change is consistent with NUREG-1431. However, the change from "bank" to "group" would result in additional rods with inoperable DRPI because the plant has 15 groups of rods and only 9 banks. Entry into this condition for 1 rod per group is acceptable because the Required ACTIONS, if applied to each nonindicating rod, provide appropriate compensatory actions. Position verification of the nonindicating rod within each group within 8 hours is adequate to allow continued full POWER OPERATION since this frequent monitoring ensures that the probability of having a rod significantly out of position and the simultaneous occurrence of an event sensitive to rod position is small.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed modification involves ACTIONS to be taken if DRPIs are inoperable. It does not involve changes in operating conditions or hardware changes that would affect the probability of an accident occurring. Monitoring rod positions and taking corrective actions are not a precursor to or assumed to be an initiator of any analyzed accidents.

The required actions to be taken if DRPIs are inoperable would ensure that the position of rods are within assumed limits or the reactor would be shutdown. Therefore, the proposed changes would not result in a significant increase in the consequences of accidents that rely on rods being within assumed limits.

Therefore, the proposed changes would not result in a significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed modification involves actions to be taken if DRPIs are inoperable. It does not involve changes in operating conditions or hardware changes that would result in plant transients or other perturbations that may result in an accident occurring. The changes do not involve a physical alteration or installation of new equipment or changes to parameters used to operate the plant. Therefore, this proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS15 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change would allow a greater number of rods to have inoperable position indication. However, each of the rods with inoperable indication would be subject to alternative monitoring methods to assure that its position was within limits. In addition, normal plant operations do not involve excessive rod movement, and the frequency that nonindicating rod position is verified ensures that rod position would be maintained within limits. There are no changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_o, F_{DH}^N , LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the foregoing information, it has been determined that the proposed changes associated with NSHC "LS15" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The ACTION statement would be modified to reflect that the requirement to initiate boration would be within "15 minutes." This change is consistent with NUREG-1431. This specific time replaces the term "immediately." This is meant to clearly specify a completed ACTION of "initiate." The requirement can be met only if boration is already taking place at 15 minutes. This time period provides adequate time for the operator to evaluate, align, and start the required systems.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The accident analysis assumes adequate minimum SDM as an initial assumption. The immediate initiation of boration is an action used to provide early restoration of minimum SDM. The action of restoring SDM can not effect the probability of an accident. The difference between "immediately" and the more defined, "within 15 minutes" may represent an increase in time to restoration of SDM but this increase has a negligible impact on the probability or consequences of an accident. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The purpose of immediate initiation of boration is to assure early restoration of minimum SDM. The slight delay in the initiation of boration does not involve any physical alternation to any plant equipment, and causes no change in the method by which any safety-related system performs its function. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety is based upon the initial assumption that the minimum SDM is met. This assumption is not effected by the insignificant increase in the time to restore minimum SDM. Therefore, there is no reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS16" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS17
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed activity would add required ACTIONS if the overall core reactivity balance was not within ± 1 percent $\Delta k/k$ of the predicted values. In the CTS, there are no Required ACTIONS; thus LCO 3.0.3 would be entered. LCO 3.0.3 requires, that within 1 hour, ACTIONS be initiated to place the plant in a condition in which the LCO did not apply. Because this particular SR is only required in MODES 1 and 2, LCO 3.0.3 would further require that the plant be placed in HOT STANDBY (MODE 3) within the following 6 hours. The proposed change, consistent with NUREG-1431, would allow 72 hours to evaluate the safety analyses and establish appropriate operating restrictions and/or SRs. If these activities were not completed within the 72 hour period, then the plant would be placed in MODE 3 within the following 6 hours.

The requirement to periodically compare the measured and predicted overall core reactivity balances provides assurance that the analytical predictions upon which the safety analyses are based accurately represent the actual core response. Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical Condition of the reactor and complete any required evaluations of the core design and safety analyses.

Following evaluations of the measurement, the core design, and the safety analysis, the cause of the reactivity anomaly may be resolved. If it is concluded that the reactor core is acceptable for continued operation, then the predicted core reactivity balance may be renormalized and Power Operation may continue. If operation restriction or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The Required Completion time of 72 hours is adequate for preparing whatever operating restrictions or surveillance that may be required to allow continued reactor operation.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not involve any new operating activities or hardware changes; thus the proposed changes has no effect on the probability of an accident. Although a small effect, the proposed change may slightly reduce the probability of an accident by allowing additional time to resolve discrepancies, and thus avoid an unnecessary plant transient (shutdown).

Satisfaction of the SR acceptance criterion provides assurance that the core-related reactivity parameters used in the safety analysis adequately represent the actual core conditions. During the 72 hour action time following an initial failure of the SR acceptance criterion, ACTIONS are established which would ensure continued agreement between the safety analysis and the actual core conditions; thereby, maintaining the validity of the safety analyses. Therefore, there is no effect on the consequences of an accident previously evaluated.

Because the available time is increased from 1 to 72 hours, the probability of an accident occurring during the time period when the plant condition is under review is slightly increased; however, the increase is small and has been previously found to be acceptable by the NRC staff through the approval of NUREG-1431.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Operation for a period of time with a discrepancy between the measured and predicted core reactivity balances is allowed by the CTS; therefore, there is no possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The accident analyses are assumed to be initiated from conditions which are consistent with the TS LCO. Thus, for the purposes of the accident analyses, it assumed that the agreement between the predicted and measured core reactivity balance is within an acceptable range. These assumptions remain valid since there is no design, operation, maintenance, or testing revision associated with this change. Therefore, there is no significant reduction in margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS17" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS20 10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed activity would add additional Required ACTIONS, which would be applicable to situations where more than 1 DRPI per bank is inoperable. The new Required ACTIONS would avoid unnecessary plant shutdowns per TS 3.0.3 yet maintain constancy with the overall protection afforded by related specifications. For example, related specifications associated with known control rod misalignments do not invoke compensatory measures as restrictive as TS 3.0.3. The situation addressed by this proposed change is one in which some misalignments or other OPERABILITY concerns, sufficient compensatory actions may be taken during the proposed 24 hour AOT in order to maintain an adequate level of plant safety.

DCPP has 53 full length rod cluster control assemblies (RCCAs) arranged in four control banks and four shutdown banks. With the exception of shutdown banks C and D, each bank is divided into two groups. Each group, in turn consists of several RCCAs which move together. Each RCCA has a position indicator channel which displays the position of the assembly. The indication of RCCA position, in accordance with Regulatory Guide (RG) 1.97, is a Category 3 variable (i.e., non-Class 1E, performance grade). Fully inserted RCCAs are further indicated by a rod at bottom signal which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

RCCAs move in preselected banks, and the banks are moved in the same preselected sequence. In the banks with two groups, the rods comprising a group operate in parallel through multiplexing thyristors. The two groups in these banks move sequentially such that the first group is within one step of the second group in the bank. A defined sequence of actuation (and deactuation) of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism.

Two separate systems are provided to sense and display control rod positions as described below:

a. DRPI system

The DRPI system measures the actual position of each control rod, using a detector which consists of discrete coils mounted concentrically with the rod drive pressure housing. The coils are located axially along the pressure housing and magnetically sense the entry and presence of the rod drive shaft through its centerline. For each detector, the coils are interlaced into two data channels of information, the DRPI system can continue to function (at reduced accuracy) when one channel fails. Multiplexing is used to transmit the digital position signals from the containment electronics to the control board display unit.

The control board display unit contains a column of light emitting diode (LEDs) for each rod. At any given time, the one LED illuminated in each column show the position for that particular rod. Each rod has its position displayed to ± 4 steps throughout its range of travel from rod bottom to 228 steps.

Included in the system is a rod at the bottom signal for each rod that operates a local alarm. Also, a control room annunciator is actuated when any shutdown rod or control bank A rod is at bottom.

b. Demand Position System

The demand position system counts pulses generated in the rod drive control system to provide a digital readout of the demanded bank position.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS20 (Continued)

The position and DRPI systems are separate systems as a result of operations requirements (i.e., no safety criteria were involved in the separation), operating procedures required the reactor operator to compare the demand and indicated (actual) readings from the DRPI system so as to verify operation of the rod control system.

The proposed Required ACTION has an AOT of 24 hours as well as compensatory actions to use the movable incore detectors to ascertain rod position, to monitor and record RCS temperature, to place rod control in the manual mode, and to limit rod motion to the extent possible via the use of other reactivity control mechanisms such as boration and dilution. The 24 hour AOT provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with an unnecessary shutdown. Monitoring and recording RCS temperature would allow early detection of mispositioned dropped rods. Overall plant safety would be enhanced by maintaining steady-state operation, as compared with the large rod movements, and potential challenges, required during an unnecessary shutdown in conduction with the loss of DRPI. The proposed Required ACTION is consistent with the overall protection afforded by the related specifications, in that acceptable power distribution limits are maintained, the minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analysis are limited.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Because no design changes are involved with this proposed change, the impact on the plant safety analysis design basis would involve a reactivity transient induced by operator error associated with the loss of position indications. The analysis results for the events in FSAR Section 15.4.1 to 15.4.3 are not dependent upon operators acts of omission or commission. The assumed reactivity insertion rates are based on conservative, worst case scenarios independent of whether they are due to equipment malfunction or human error. Loss of RCCA position indication would not affect the assumed reactivity insertion rates. Further, the protection systems assumed in the analyses of these events are unaffected by the proposed change.

The worst case reactivity transient of this nature (the withdrawal of a single RCCA) has been analyzed in FSAR Section 15.4.3 assuming that the operators ignore RCCA position indication. Whether indication is lost, as is the case covered by the proposed ACTION statement, or disregarded, does not change the method of analysis or the results of the analysis.

The accident analysis are initiated from within the conditions defined by the TS LCO and they remain unchanged; therefore, the accident analyses are unaffected. Therefore the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS20 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiator are introduced by the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The accident analyses are assumed to be initiated from conditions which are consistent with the TS LCO. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no affect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS20" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS21
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would revise the shutdown requirement for control rod banks not within limits from "Be in at least HOT STANDBY within 6 hours" to "Be in at least HOT STANDBY within the next 6 hours." This change would provide additional time to place the plant in MODE 3 because the 6 hour limit would not commence until after the time periods allowed for completing other actions, such as restoring the control bank position within 2 hours. The CTS require the 6 hour period to commence at the time the LCO is not met. The proposed change is in accordance with NUREG-1431.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The control rod insertion LCO is intended to assure that the control rods are in position to mitigate accidents or transients that depend on a reactor trip and that reactor power distribution is in accordance with the assumptions used in accident analyses. If control rods are outside of specified limits, the plant should be placed in a MODE for which the LCO is not applicable, i.e., MODE 3. As noted in the Bases of NUREG-1431, 8 hours is a reasonable time, based on operating experience, for reaching the required MODE from full power Conditions, in an orderly manner without challenging plant systems.

The proposed shutdown requirement Completion Time change would result in an extension of time to achieve MODE 3 from 6 hours (per CTS) to a maximum of 8 hours (2 hours to attempt to complete other mitigating actions followed by 6 hours to achieve MODE 3). The proposed change would not change the plant design or operations such that an accident or transient could be initiated. By allowing a shutdown time based on operating experience, the change would reduce the chances of an operator error or challenge to plant systems that could result from the more restrictive requirements in the CTS. Thus, the change would have no adverse effect on the probability of occurrence of an accident.

The proposed change would not affect the method of operation of plant systems and involves only the time requirement to achieve a reactor shutdown when the control rods are not within limits. The probability that an accident would occur during the 2 hour time extension allowed by the proposed change would be negligible. Also, during the 2 hour period, the other Required ACTIONS are being performed. These include returning control rods to within limits, restoration or SDM as necessary, and reduction in reactor power. Therefore, as time in the ACTION statement passes, the plant is being brought closer to Conditions for which the accident assumptions are valid. Thus, the proposed change would have a negligible effect on the consequences of accidents previously analyzed.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS21 (Continued)

Therefore, the proposed change would not result in a significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Operation in accordance with the proposed change would not introduce any new failure modes for plant systems and components. Only the duration of operation in the ACTION statement is affected. The proposed change is administrative in nature and does not involve any physical alteration to any plant equipment, and causes no change in the method by which any safety-related system performs its function. Therefore, this proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The margins of safety involved with this proposed change are those associated with accidents that rely on control rod position to assure reactor trip effectiveness and to assure power distribution limits are maintained. When the rod position is not within limits, the LCO initiates the corrective action to restore the margins by specifying mitigating actions. While the margins of safety may be affected by failure to meet the LCO, the additional 2 hours to achieve MODE 3 allowed by the proposed change has no effect on them. The control rod insertion limits are not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_0, F_{DH}^N , LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-21" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS22
10 CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The current SR requiring shutdown rod insertion to be verified within 15 minutes prior to withdrawing control rods during approaches to criticality would be deleted. This requirement has essentially been incorporated into the ITS LCO 3.1.5 Applicability which would require that the shutdown banks meet the LCO when in MODE 2 with any control bank not fully inserted. The requirement to perform this check prior to entering the LCO Applicability is covered in plant startup procedures which require the operator to closely monitor rod position.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92 (c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Under the CTS, shutdown rod insertion is required to be verified within limit (withdrawn to at least [225] steps) within 15 minutes prior to control rod withdrawal. Since the specification has no CTS 4.0.4 exception, shutdown rod insertion must be within limit prior to entering the LCO Applicability. The current SR requires an arbitrary estimate of when the plant is 15 minutes away from control rod withdrawal. This has no basis from the accident analyses, which are satisfied as long as the shutdown rods satisfy their insertion limit prior to control rod withdrawal.

Based on the foregoing discussion, the proposed change would not involve a significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

This change will not impact plant system operations or other conditions that could cause an accident of a new or different type. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS22 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The shutdown rod insertion limit is not changed nor are there are changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_Q, F_{QH}^N , LOCA PCT, peak local power density, or any other margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-22" resulting from the conversion to the ITS format satisfy the NSCH standards of 10 CFR 50.92(c), and accordingly a no NSCH finding is justified.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR2 10CFR50.92 EVALUATION FOR RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change in accordance with NUREG-1431, Rev. 1, removes the requirement for a special report to be generated and submitted to the NRC. Reporting to the NRC will be done commensurate with the reporting requirements of 10CFR50.72 and 50.73.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92 (c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change is purely an administrative reporting change and cannot affect any accident probability or consequences.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

This change is purely an administrative reporting change and cannot create any new accident or affect any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change is purely an administrative reporting change and does not affect any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "TR-2" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92(c), and does not involve a significant hazards consideration.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR3
10 CFR 50.92 EVALUATION
FOR
RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN TECHNICAL SPECIFICATIONS

This proposed revision is to remove reference to specific post-maintenance tests from the CTS. Post-maintenance testing programs are controlled via plant administrative procedures in accordance with licensee controlled document (ITS Section 5.4.1 commitments to NRC RG 1.33, "Quality Assurance Program Requirements Operation" and ANS 3.2-ANSI N18.7, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants)." Specific post-maintenance testing requirements are contingent on the type and scope of maintenance actually performed as well as the availability and viability of test equipment, techniques, etc. Removal of specific testing requirements from the CTS and reliance on normal post-maintenance testing programs addressed by licensee controlled documents allow flexibility to modify testing to address the circumstances of the maintenance performed while still assuring OPERABILITY of equipment returned to service.

This proposed TS change has been evaluated and it has been determined that it involves NSCH. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92 (c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This is an administrative change which removes specific post-maintenance test requirements from the CTS. The testing, or equivalent testing, to assure equipment OPERABILITY prior to return to service would still be done as required by normal plant maintenance retest programs. Therefore, this change would not result in any increase in the probability or consequences of an accident previously evaluated.
2. Does the change create the possibility of a new or different kind of accident from any previously evaluated?

This is an administrative change and does not create a new or different kind of accident from any previously evaluated.
3. Does this change involve a significant reduction in margin of safety?

This is an administrative change and does not affect any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed change associated with NSHC "TR-3" does not meet the requirements of 10 CFR 50.92© and does not involve a significant hazards consideration satisfies the NSHC standards of 10 CFR 50.92(c), and accordingly a NSCH finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers (1 Page)

NUREG-1431 Specifications that are not applicable (1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
3.1.1	3.1-1
3.1.2	Not Used
3.1.3 . (Re-numbered to 3.1.2)	3.1-2
3.1.4 . (Re-numbered to 3.1.3)	3.1-4
3.1.5 . (Re-numbered to 3.1.4)	3.1-7
3.1.6 . (Re-numbered to 3.1.5)	3.1-10
3.1.7 . (Re-numbered to 3.1.6)	3.1-12
3.1.8 . (Re-numbered to 3.1.7)	3.1-14
3.1.9	Not Used
3.1.10 (Re-numbered to 3.1.8)	Note (1) .. 3.1-17
3.1.11	Not Used

Methodology (2 Pages)

Note (1): See conversion for TS 3/4.10

Industry Travelers Applicable to Section 3.1

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated	3.1-1	NRC approved.
TSTF-12, Rev. 1	Incorporated	3.1-15	NRC approved. ITS Special Test Exception 3.1.10 is retained and re-numbered as 3.8.1, consistent with this traveler and TSTF-136.
TSTF-13, Rev. 1	Incorporated	3.1-4	NRC approved.
TSTF-14, Rev. 3	Incorporated	3.1-13	NRC approved.
TSTF-15, Rev. 1	Incorporated	N/A	NRC approved.
TSTF-89	Incorporated	3.1-8	NRC approved.
TSTF-107	Incorporated	3.1-6	
TSTF-108, Rev. 1	Not Incorporated	N/A	Not NRC approved as of cut-off date.
TSTF-110, Rev. 1	Incorporated	3.1-10	
TSTF-136	Incorporated	3.1-9, 3.1-15	
TSTF-141	Not Incorporated	N/A	Disagree with change; traveler issued after cut-off date
TSTF-142	Not Incorporated	N/A	Traveler issued after cut-off date.
WOG-73, Rev. 1	Incorporated	3.1-7	
WOG-105	Incorporated	3.1-16	

NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

Specification #	Specification Title	Comments
3.1.2	SHUTDOWN MARGIN (SDM) - $T_{avg} \leq 200^{\circ}\text{F}$	Incorporated into ITS 3.1.1 per Traveler TSTF-136.
3.1.9	PHYSICS TESTS - MODE 1 Test Exceptions	This specification was deleted and ITS 3.1.10 was reabeled to be 3.1.8 per TSTF-12, Rev. 1 and TSTF- 136.
3.1.11	SHUTDOWN MARGIN (SDM) - Test Exceptions	This specification was deleted per TSTF-12, Rev. 1.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM) — $T_{avg} > 200^{\circ}\text{F}$

LCO 3.1.1 SDM shall be $\geq 1.6 \Delta k/k$ within the limits provided in the COLR.

APPLICABILITY: MODE 2 with $k_{eff} < 1.0$,
 MODES 3, 4, and 5

NOTE: While this LCO is not met, entry into MODE 5 from MODE 6 is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM is $\geq 1.6 \Delta k/k$ to be within limits	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

~~3.1.3~~ ~~3.1.2~~ Core Reactivity

LCO ~~3.1.3~~ ~~3.1.2~~ The measured core reactivity shall be within $\pm 1\%$ $\Delta k/k$ of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	72 hours
	<u>AND</u>	
	A.2 Establish appropriate operating restrictions and SRs.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.3.1 3.1.2.1 -----NOTE-----</p> <p>The predicted reactivity values may shall be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading.</p> <p>-----</p> <p>Verify measured core reactivity is within $\pm 1\% \Delta k/k$ of predicted values.</p>	<p style="text-align: right;"><u>3.1-2</u></p> <p>Once prior to entering MODE 1 after each refueling</p> <p><u>AND</u></p> <p>-----NOTE-----</p> <p>Only required after 60 EFPD</p> <p>-----</p> <p>31 EFPD thereafter</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 3.1.3 Moderator Temperature Coefficient (MTC)

LC0 3.1.4 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be $\leq \Delta k/k^{\circ}F$ at hot zero power that specified in Figure 3.1.3-1.

B

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit, MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within upper limit.	A.1 Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2 with $k_{eff} < 1.0$.	6 hours
C. MTC not within lower limit.	C.1 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.4.1 3.1.3.1 Verify MTC is within upper limit.</p>	<p>Once prior to entering MODE 1 after each refueling</p>
<p>SR 3.1.4.2 Verify MTC is within 300 ppm Surveillance limit specified in the COLR.</p>	<p>NOTE Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm Once each cycle</p>
<p>SR 3.1.4.3 3.1.3.2 -----NOTES-----</p> <p>1. Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm</p> <p>1-2. If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.4-3.2 shall be repeated once per 14 EFPD during the remainder of the fuel cycle.</p> <p>2-3. SR 3.1.4-3.2 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of ≤ 60 ppm is less negative than the 60 ppm Surveillance limit specified in the COLR.</p> <p>-----</p> <p>Verify MTC is within lower limit.</p>	<p>NOTE Not required to be performed until 7 EFPD after reaching the equivalent of an equilibrium RTP ARO boron concentration of 300 ppm Once each cycle</p>

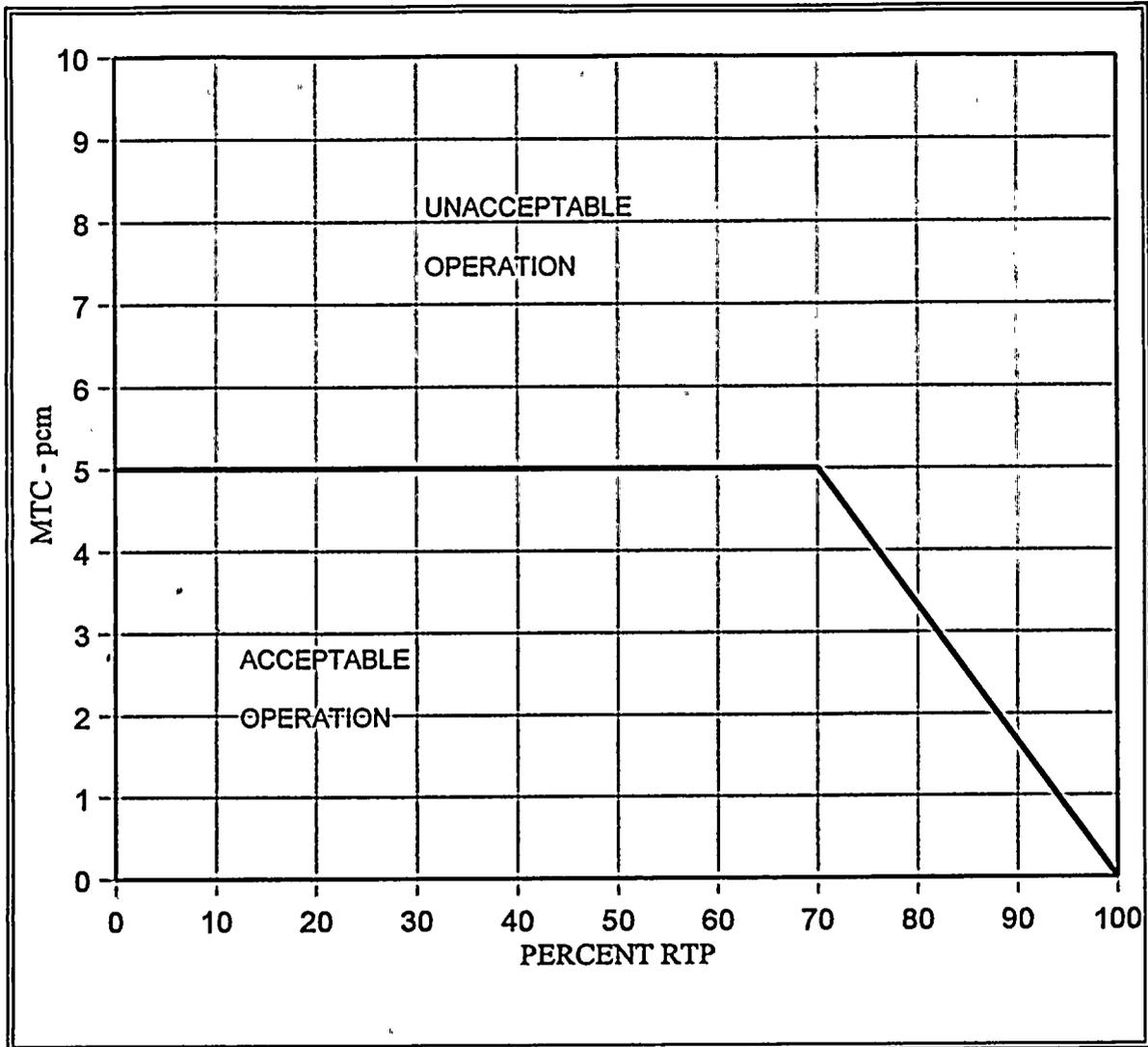


FIGURE 3.1.3-1 (page 1 of 1) MODERA

TOR TEMPERATURE COEFFICIENT vs. POWER LEVEL

3.1 REACTIVITY CONTROL SYSTEMS

~~3.1.5~~ ~~3.1.4~~ Rod Group Alignment Limits

LCO ~~3.1.5~~ ~~3.1.4~~ All shutdown and control rods shall be OPERABLE. with all

3.1-5

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

3.1-6

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) untrippable-inoperable	A.1.1 Verify SDM is > 1.6 % $\Delta k/k$ to be within the limits provided in the COLR	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Be in MODE 3.	6 hours
B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits.	1 hour
	<u>OR</u>	
	B.2.1.1 Verify SDM is > 1.6 % $\Delta k/k$ to be within the limits provided in the COLR	1 hour
	<u>OR</u>	(continued)

3.1-1
3.1-6

3.1-1

CONDITION	REQUIRED ACTION	COMPLETION TIME	
B. (continued)	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour	
	<u>AND</u>		
	B.2.2 Reduce THERMAL POWER to $\leq 75\%$ RTP.	2 hours	
	<u>AND</u>		
	B.2.3 Verify SDM is > 1.6 % Ak/k to be within the limits provided in the COLR.	Once per 12 hours	<u>3.1-1</u>
	<u>AND</u>		
C. Required Action and associated Completion Time of Condition B not met.	B.2.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	72 hours	
	<u>AND</u>		
	B.2.5 Perform SR 3.2.2.1.	72 hours	<u>3.1-16</u>
	<u>AND</u>		
D. More than one rod not within alignment limit.	B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days	
	C.1 Be in MODE 3.	6 hours	
	<u>OR</u>		
D.1.1 Verify SDM is > 1.6 % Ak/k to be within the limits provided in the COLR.	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour	<u>3.1-1</u>
	<u>AND</u>		
	D.2 Be in MODE 3.	6 hours	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 3.1.4.1 Verify individual rod positions within alignment limit.	12 hours AND Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable
SR 3.1.5.2 3.1.4.2 Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days
SR 3.1.5.3 3.1.4.3 Verify rod drop time of each rod, from the fully withdrawn position, is \leq 2.2 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500^{\circ}\text{F}$; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

3.1-10

B-PS

3.1 REACTIVITY CONTROL SYSTEMS

~~3.1.6~~ ~~3.1.5~~ Shutdown Bank Insertion Limits

LCO ~~3.1.6~~ ~~3.1.5~~ Each shutdown bank shall be within insertion limits specified in the COLR.

APPLICABILITY: MODE 1,
 MODE 2 with any control bank not fully inserted.

-----NOTE-----
 This LCO is not applicable while performing SR ~~3.1.5.2~~ ~~3.1.4.2~~.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1 Verify SDM is > 1.6 % Ak/k to be within the limits provided in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore shutdown banks to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.6.1 3.1.5.1 Verify each shutdown bank is within the limits specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 3.1.6 Control Bank Insertion Limits

LCO 3.1.7 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

-----NOTE-----

This LCO is not applicable while performing SR 3.1.5.2 3.1.4.2.

3.1-9

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control bank insertion limits not met.	A.1.1 Verify SDM is ≥ 1.6 % k_{eff} to be within the limits provided in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
B. Control bank sequence or overlap limits not met.	A.2 Restore control bank(s) to within limits.	2 hours
	B.1.1 Verify SDM is ≥ 1.6 % k_{eff} to be within the limits provided in the COLR.	1 hour
	<u>OR</u>	
	B.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	B.2 Restore control bank sequence and overlap to within limits.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

3.1-1

3.1-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 3.1.6.1 Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality
SR 3.1.7.2 3.1.6.2 Verify each control bank insertion is within the limits specified in the COLR.	12 hours AND Once within 4 hours and every 4 hours thereafter when the rod insertion limit monitor is inoperable
SR 3.1.7.3 3.1.6.3 Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1-8 3.1-7 Rod Position Indication

LCO 3.1-8 3.1-7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

B-PS

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
 Separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank.

3.1-7

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors. OR A.2 Reduce THERMAL POWER to \leq 50% RTP.	Once per 8 hours 8 hours
B. More than one DRPI per group inoperable.	B.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors. AND B.2 Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	Once per 8 hours 24 hours
B C. One or more rods with inoperable position indicators DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	BC.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors. OR BC.2 Reduce THERMAL POWER to \leq 50% RTP.	4 hours 8 hours

B-PS

3.1-12

3.1-7

3.1-12

3.1-17

B

3.1-12

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1-8.1 3.1-7.1 Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	18 months Once prior to criticality after each removal of the reactor vessel head

3.1 REACTIVITY CONTROL SYSTEMS

~~3.1.10~~ ~~3.1.8~~ PHYSICS TESTS Exceptions - MODE 2

LCO ~~3.1.10~~ ~~3.1.8~~ During the performance of PHYSICS TESTS, the requirements of

- LCO ~~3.1.43~~ 1.3, "Moderator Temperature Coefficient (MTC)";
 - LCO ~~3.1.53~~ 1.4, "Rod Group Alignment Limits";
 - LCO ~~3.1.63~~ 1.5, "Shutdown Bank Insertion Limits";
 - LCO ~~3.1.73~~ 1.6, "Control Bank Insertion Limits"; and
 - LCO 3.4.2; "RCS Minimum Temperature for Criticality"
- may be suspended, provided:

- a. RCS lowest ~~operating~~ loop average temperature is $\geq 531^\circ\text{F}$; and
- b. SDM is $\geq 1.6\% \Delta k/k$ within the limits provided in the ~~COLR~~ and
- c. ~~THERMAL POWER is $\leq 5\%$ RTP~~

3.1-9
3.1-20
B-PS
3.1-1
3.1-13

APPLICABILITY: ~~MODE 2-d~~ During PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit. AND A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
C. RCS lowest operating loop average temperature not within limit.	C.1 Restore RCS lowest operating loop average temperature to within limit.	15 minutes
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.10.1 8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1.	Within 12 hours prior to initiation of PHYSICS TESTS
SR 3.1.10.2 8.2	Verify the RCS lowest operating loop average temperature is $\geq 531^{\circ}\text{F}$.	30 minutes
SR 3.1.10.3 8.3	Verify THERMAL POWER is $\leq 5\%$ RTP	1 hour
SR 3.1.10.4 8.4	Verify SDM is $\geq 1.6\%$ Ak/k, within the limits provided in the COLR	24 hours

B
3.1-20
B
3.1-13
3.1-1

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions -** The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions -** The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications -** The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes-** Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts -** The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

**Methodology For Mark-up of NUREG-1431 Specifications
(Continued)**

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES CONTENTS

Mark-up:

<u>BASES</u>	<u>PAGE</u>
3.1.1	B 3.1-1
3.1.2	Not Used
3.1.3 (Re-numbered to 3.1.2)	B 3.1-5
3.1.4 (Re-numbered to 3.1.3)	B 3.1-10
3.1.5 (Re-numbered to 3.1.4)	B 3.1-14
3.1.6 (Re-numbered to 3.1.5)	B 3.1-22
3.1.7 (Re-numbered to 3.1.6)	B 3.1-24
3.1.8 (Re-numbered to 3.1.7)	B 3.1-28
3.1.9	Not Used
3.1.10 (Re-numbered to 3.1.8) Note (1)	B 3.1-33
Methodology	(1 Page)

Note (1): See conversion for TS Section 3/4.10

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Rod Control Rod System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Rod Control Rod System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The soluble boron Chemical and Volume Control System can control the soluble boron concentration to compensate for fuel depletion during operation and all xenon burnout reactivity changes and can maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured, assuming that core reactivity is within design limit of LCO 3.1.2, by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.76, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration

APPLICABLE
SAFETY ANALYSIS

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are not exceeded. This is done by ensuring that:

- a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;
- b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and $\leq 280\ 200$ cal/gm energy deposition average fuel pellet enthalpy at the hot spot in irradiated fuel for the rod ejection accident, Ref. 5); and
- c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accidents for the SDM requirements ~~is based on a~~ are the main steam line break (MSLB) and inadvertent boron dilution accidents, as described in the accident

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

analysis FSAR (Refs. 2 and 3). In addition to the limiting MSLB transient, the SDM requirement is also used in the analyses of the following events:

- a. Inadvertent boron dilution;
- b. An uncontrolled rod withdrawal from subcritical or low power condition;
- c. Start of an inactive reactor coolant pump (RCP); and
- d. Rod ejection.

The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life with RCS T_{avg} equal to 547°F. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

~~In addition to the limiting MSLB transient, the SDM requirement is also used in the analyses of the following events:~~

- ~~a. Inadvertent boron dilution;~~
- ~~b. An uncontrolled rod withdrawal from subcritical or low power condition; and~~
- ~~c. Start of an inactive reactor coolant pump (RCP); and~~
- ~~d. Rod ejection.~~

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

The startup of an inactive RCP in MODES 1 or 2 is precluded. In MODE 3, the startup of an inactive RCP can not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent start is less than half the minimum required

(Continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

SDM satisfies Criterion 2 of the 10CFR50.36(c)(2)(iii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumption.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable sufficient. The required SDM is specified in the COLR.

APPLICABILITY

In MODE 2 with $k_{\text{eff}} < 1.0$ and in MODES 3, and 4, and 5 the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. ~~In MODE 5, SDM is addressed by LCO 3.1.2, "SHUTDOWN MARGIN (SDM) $T_{\text{SDM}} \leq 200^{\circ}\text{F.}$ "~~ In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.65, "Shutdown Bank Insertion Limits," and LCO 3.1.76, "Control Bank Insertion Limits"

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the borated water source should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

BASES

~~In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% $\Delta k/k$ must be recovered and a boration flow rate of [] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of the RCS~~

SURVEILLANCE
REQUIREMENTSSR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.65 and LCO 3.1.76 are met. In the event that a rod is known to be untrippable, however, SDM

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(Continued)

verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODES 2 (with $k_{eff} < 1.0$), 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects (SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth):

- a. RCS boron concentration;
- b. Control and shutdown rod position;
- c. RCS average temperature;
- d. Fuel burnup based on gross thermal energy generation;
- e. Xenon concentration;
- f. Samarium concentration; and
- g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Chapter 15, Section 15.4.2.1
3. FSAR, Chapter 15, Section 15.2.4.
4. 10 CFR 100.
5. FSAR, Chapter 15, Section 15.4.6.1.6

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 3.1.2 Core Reactivity

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM) $\rightarrow 200^{\circ}\text{F}$ ") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

(Continued)

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluate.

APPLICABLE
SAFETY ANALYSIS

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant SAFETY ANALYSES operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to

(Continued)

accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified

(Continued)

APPLICABLE
SAFETY ANALYSIS
(continued)

against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide (continued) an accurate representation of the core reactivity.

Design calculations are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion as well as providing inputs to the safety analysis.

The comparison between measured and predicted initial core reactivity provides a validation of the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after when deemed necessary shall be performed after reaching RIP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of 10CFR50.36(c)(2)(11).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily altered once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of $\pm 1\% \Delta k/k$ has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

(Continued)

When measured core reactivity is within 1% $\Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel

(Continued)

movements are performed within the bounds of the safety analysis. An SDM demonstration is Core reactivity and control rod worth measurements are required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve and the boron concentration requirement for SDM may be renormalized and power operation may continue. If operational restrictions or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing whatever operating restrictions or surveillances that may be required to allow continued reactor operation.

(Continued)

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by SR 3-1-1.1 LCO 3-1-1 Required Action A.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant system.

SURVEILLANCE
REQUIREMENTS

SR 3-1-3-1 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note. The Note indicates that the normalization (adjustment, only if necessary) of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial

(Continued)

60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

- REFERENCES
1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
 2. FSAR, Chapter 15.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both ~~initial and~~ Reload cores are designed so that the beginning of cycle (BOC) MTC is less than zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the FSAR accident and transient analyses.

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

APPLICABLE
SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

- a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and
- b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The FSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

(Continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(Ref. 2 (Continued))

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodged and unrodged conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches a boron concentration equivalent to 300 ppm at an equilibrium, all rods out, RTP condition. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC) was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR accident analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) adding margin to this value to account for the largest difference in MTC observed between an EOC, all rods withdrawn, RATED THERMAL POWER condition and an envelope of those most adverse conditions of moderator temperature and pressure, rods inserted to their insertion limits, axial power skewing, and xenon concentration that can occur in normal operation within technical specification limits and lead to a significantly more negative EOC MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR accident analyses into the limiting EOC MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by adding an allowance for burnup and soluble boron concentration changes to the limiting EOC MTC value.

(Continued)

MTC satisfies Criterion 2 of the 10CFR50.36(c)(2)(iii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.43 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive near BOC when core reactivity and required boron concentration are at their maximum values; this upper bound must not be exceeded. This maximum upper limit is evaluated near BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive limit and the EOC negative limit are established in the COLR to allow specifying limits for

(Continued)

BASES

LCO
(Continued)

each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled control rod assembly or group withdrawal) will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODE.

ACTIONS

A.1

If the upper MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time in cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits are not established within 24 hours, the unit must be brought to MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC that is more positive than that assumed in safety analyses.

(Continued)

BASES

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the EOC MTC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the EOC MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

(Continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.1.4.1 3.1.3.1

This SR requires measurement of the MTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the BOC limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOC MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

SR 3.1.4.2 3.1.3.2 and SR 3.1.4.3

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.4.3 2 is modified by a ~~three~~ Notes that include the following requirements:

1. The SR is required to be performed once each cycle within 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.

a2. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 EFPD is sufficient to avoid exceeding the EOC limit.

(Continued)

BASES

b3. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is less negative than the 60 ppm Surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnu.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
 2. FSAR, Chapter 15.
 3. WCAP-9273-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
 4. ~~FSAR, Chapter [15].~~
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 3.1.4

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately $\frac{1}{8}$ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3-1.5 3-1.4

BASES

The RCCAs are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. All units have four control banks and at least two shutdown banks. Control banks contain two rod groups. Two shutdown banks (A and B) contain two rod groups and the remaining two shutdown banks (C and D) contain one rod group.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System. The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a

(Continued)

BASES

BACKGROUND
position (continued)

group all receive the same signal to move and should, therefore, all be at the same indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or $\pm \frac{5}{8}$ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is ± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches. The DRPI system is capable of monitoring rod position within at least ± 12 steps with either full accuracy or half accuracy.

APPLICABLE
SAFETY ANALYSIS

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing rod inoperability or misalignment are that:

- a. There be no violations of:
 1. Specified acceptable fuel design limits, or
 2. Reactor Coolant System (RCS) pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control or shutdown rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

(Continued)

BASES

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 4).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned rod is allowed if the heat flux hot channel factor ($F_0(Z)$) and the nuclear enthalpy hot channel factor (F_{AH}^N) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod

(Continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_0(Z)$ and $F_{\Delta H}^N$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_0(Z)$ and $F_{\Delta H}^N$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO /

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability to meet SDM) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required time assumed in the safety analyses. Rod control malfunctions that result in the inability to move a rod (e.g., rod urgent failures), but do not impact trippability, do not necessarily result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

The requirement to maintain rod alignment is met by comparing individual rod DRPI indication and bank demand position indication to be within plus or minus 12 steps. If one of these position indicators become inoperable, the conditions of this LCO are still met by compliance with LCO 3.1.7.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

(Continued)

BASES

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are typically fully inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM) $\geq 200^\circ\text{F}$," for SDM in MODES 2 with $K_{eff} < 1.0$, 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refuelin.

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

With an inoperable rod(s), this ACTION provides for verification of SDM; this is most simply accomplished by verifying rod insertion limits are met. Additionally, actions could include calculation of the current SDM and boration to meet limits specified in the COR or proceed

(Continued)

BASES

ACTION
(continued)

to MODE 3. These actions are consistent with those specified in LCO 3.1.5 and LCO 3.1.6.

A rod is considered trippable if it was demonstrated OPERABLE during the last performance of SR 3.1.4.2 and met the rod drop time criteria during the last performance of SR 3.1.4.3.

In this situation, SDM verification must account for the absence of the negative reactivity of the untrippable rod(s), as well as the rod of maximum worth.

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable (i.e., OPERABLE). If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group demand position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.6 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.7 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

(Continued)

BASES

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be fully inserted and control bank C must be inserted to approximately 100 steps to 115 steps.

Power operation may continue with one RCCA OPERABLE (i.e. trippable) but misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_0(Z)$ and $F_{\Delta H}^N$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible. Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 3). The Completion

(Continued)

BASES

ACTION
 reduction
 (continued)

Time of 2 hours gives the operator sufficient time to accomplish an orderly power without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_0(Z)$ and $F_{\Delta H}^N$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_0(Z)$ and $F_{\Delta H}^N$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

The following accident analyses require re-evaluation for continued operation with a misaligned rod:

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

(Continued)

BASES

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group-average demand position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to

(Continued)

BASES

ACTIONS
and (continued)

complete the action. This allows the operator sufficient time to align the required valves start the boric acid pumps. Boration will continue until the required SDM is restored.

Additionally, the requirements of LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," apply if the misaligned rods are not within the required insertion limits.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant system.

SURVEILLANCE
REQUIREMENTS

SR 3.1.5.1 3.1.4

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. ~~If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal.~~ The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

(Continued)

BASES

SR 3-1-5-2 3-1-4-2

Verifying each rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each rod would result in radial or axial power tilts, or oscillations. Exercising each individual rod every 92 days provides confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3-1-5-1 3-1-4-1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between or during required performances of SR 3-1-5-2 3-1-4-2 (determination of rod OPERABILITY by movement), if a rod(s) is discovered to be immovable, but remains trippable, the rod(s) is considered to be OPERABLE. At any time, if a rod(s) is immovable, a determination of the trippability (OPERABILITY) of the rod(s) must be made, and appropriate action taken.

SR 3-1-5-3 3-1-4-3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature $\geq 500^{\circ}\text{F}$ to simulate a reactor trip under actual conditions.

(Continued)

BASES

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
 2. 10 CFR 50.46.
 3. FSAR, Chapter 15, Section 15.2.3.
 4. FSAR, Chapter 15, Section 4.2.3
 5. ~~FSAR, Chapter [15]~~
 6. ~~FSAR, Chapter [15]~~
 7. ~~FSAR, Chapter [15]~~
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3-1.6 3-1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. ~~All plants have four control banks and at least two shutdown banks.~~ ~~four control banks contain two rod groups. Two shutdown banks (A and B) contain two rod groups and the remaining two shutdown banks (C and D) consist of a single group.~~ See LCO 3-1.6 3-1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3-1.8 3-1.7, "Rod Position Indication," for position indication requirements.

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 3.1.5 Shutdown Bank Insertion Limits

BASES

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE
SAFETY ANALYSIS

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.7 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN - (SDM) $T_{avg} > 200^{\circ}\text{F}$," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) $T_{avg} < 200^{\circ}\text{F}$ ") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 ~~3.1.5~~ Shutdown Bank Insertion Limits

BASES

APPLICABLE
SAFETY ANALYSIS
(Continued)

maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion safety limits and inoperability or misalignment is that:

- a. There be no violations of:
 1. Specified acceptable fuel design limits, or
 2. RCS pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of ~~10CFR50.36(c)(2)(ii)~~.

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B ~~3.1.6~~ 3.1.5 Shutdown Bank Insertion Limits

BASES

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. The applicability in MODE 2 begins at initial control bank withdrawal, during an approach to criticality, and continues throughout MODE 2, until all control bank rods are again fully inserted by reactor trip or by shutdown. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks are typically fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 and ~~LCO 3.1.2~~ for SDM requirements in MODES 2 with ~~K_{eff} > 1.0~~ 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.5-24.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS .

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3-1-6 3-1-5 Shutdown Bank Insertion Limits

BASES

ACTIONS
(continued)

for an extended period of time. Additionally, the requirements of LCO 3-1-5 3-1-4, "Rod Group Alignment Limits," apply if one or more shutdown rods are not within the required alignment limits.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3-1-6-1 3-1-5-1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
 2. 10 CFR 50.46.
 3. FSAR, Chapter 15, Section 15.4.3.2.4
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. All plants have four control banks and at least two shutdown banks. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously and are moved in a staggered fashion, but always within one step of each other. Two shutdown banks (C and D) consist of a single group. See LCO 3.1.6 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.8 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. An example is provided for information only in Figure B 3.1.7 1. The control banks are required to be at or above the insertion limit lines.

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3-1.7 3-1.6 Control Bank Insertion Limits

BASES

Figure B 3-1.7-1 The COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance traveled together by two control banks. The predetermined position of control bank C, at which control bank D will begin to move with bank C on a withdrawal, will be at 118 steps for a fully withdrawn position of 231 steps. The fully withdrawn position is defined in the COLR. The control banks are used for precise reactivity control of the reactor. The positions of the control banks can be controlled manually, or automatically by the Rod Control System. They are capable of altering reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3-1.5 3-1.4, "Rod Group Alignment Limits," LCO 3-1.6 3-1.5, "Shutdown Bank Insertion Limits," LCO 3-1.7 3-1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained assuming LCO 3-1.2 Core Reactivity is met for core reactivity.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the

(Continued)

BASES

BACKGROUND

event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident (continued) requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE
SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

- a. There be no violations of:
 - 1. Specified acceptable fuel design limits, or
 - 2. Reactor Coolant System pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 4).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 5).

The insertion limits satisfy Criterion 2 of 10CFR50.36(c)(2)(1), in that they are initial conditions assumed in the safety analysis.

(Continued)

BASES

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.5-212. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

(Continued)

BASES

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

- a. Reducing power to be consistent with rod position; or
- b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN - (SDM) \rightarrow 200°F") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits. Failure of sequence or overlap support equipment does not require entering the ACTIONS as long as sequence and overlap limits are maintained.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

Additionally, the requirements of LCO 3.1.4, "Rod Group Alignment Limits," apply if one or more control rods are not within the required alignment limits.

(Continued)

BASES

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 3, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.7-16.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

(Continued)

BASES

SURVEILLANCE
REQUIREMENT

SR 3.1.7-26.2

Verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY and to detect control banks that may be approaching the insertion (continued) limits since, normally, very little rod motion occurs in 12 hours. ~~If the insertion limit monitor becomes inoperable, verification of the control bank position at a frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.~~

SR 3.1.7-36.3

When control banks are maintained within their insertion limits as checked by SR ~~3.1.7-2 3.1.6.2~~ above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. ~~The verification of compliance with the sequence and overlap limits specified in the COLR consists of an observation that the static rod positions of those control banks not fully withdrawn from the core are within the limits specified in the COLR. Bank sequence and overlap must also be maintained during rod movement, implicit within the LCO. A Frequency of 12 hours is consistent with the insertion limit check above in SR ~~3.1.7.2 3.1.6.2~~.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.
 2. 10 CFR 50.46.
 3. FSAR, Chapter 4, Section 4.3.2.4
 4. FSAR, Chapter 4, Section 4.3.2.4
 5. FSAR-ICAP-9273-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985
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B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1-8 3.17 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1-8 3.17 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEM

B 3-1-8 3-17 Rod Position Indication

BASES

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or $\pm \frac{5}{8}$ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual ~~control rod position~~, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of DRPI System is ~~± 6 steps (± 3.75 inches), and the maximum uncertainty is ± 12 step (± 7.5 inches).~~ With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position and the demand position could be 24 steps or 15 inches. The DRPI system is capable of monitoring rod position within at least ± 12 steps with either full accuracy or half accuracy.

(Continued)

BASES

APPLICABLE
SAFETY ANALYSIS

Control and shutdown rod position accuracy is essential during power operation. Powerpeaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the bank sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.6 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.7 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.5 3.1.4, "Rod Group Alignment Limits"). Control Rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of 10CFR50.36(c)(2)(ii). The control rod position indicators monitor control rod position, which is an initial condition of the accident.

LCO

LCO 3.1.8 3.1.7 specifies that the DRPI System and the Bank Demand Position Indication System be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

- a. The DRPI System on either full accuracy or half accuracy indicates within 12 steps of the group step counter demand position as required by LCO 3.1.5 3.1.4, "Rod Group Alignment Limits"; and
- b. The Bank Demand Indication System has been reset in the fully inserted position, fully withdrawn position or to the DRPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of control rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.5 3.1.4, in position indication for a single rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

(Continued)

BASES

These requirements ensure that ~~control~~ rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned ~~control~~ rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3-1-5 3-1-4, LCO 3-1-6 3-1-5, and LCO 3-1-7 3-1-6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator per bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

(Continued)

BASES

ACTIONS
(continued)

A.1

When one DRPI per group fails, the position of the rod may still be determined indirectly by use of the movable incore detectors. The Required Action may also be ensuring at least once per hours that F_{ij} satisfies LCO 3.2.1, F_{ij} satisfies LCO 3.2.2, and SDM is within the limits provided in the COLR, provided the nonindicating rods have not been moved. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to $\leq 50\%$ RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3).

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to $\leq 50\%$ RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1 and B.2

When more than one DRPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. The indirect position determination available via movable incore detectors will minimize the potential for rod misalignment.

(Continued)

BASES

The position of the rods can be determined indirectly by use of the movable incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_0 satisfies LCO 3.2.1, F_H satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided in the CLR, provided that the nonindicating rods have not moved. Verification of RCCA position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Allowed Outage Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with a shutdown without full rod position indication.

Based on operating experience, normal power operation does not require excessive rod movement. If one or more control rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

B.1 and B.2C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable position indicators DRPIs have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 or B.1 are still appropriate but must be initiated promptly under Required Action B.1 C.1 to begin indirectly verifying that these rods are still properly positioned, relative to their group positions.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to $\leq 50\%$ RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at $> 50\%$ RTP, if one or more rods are misaligned by

(Continued)

BASES

more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions using the movable incore detectors.

C.1.1 and C.1.2D.1.1 and D.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are

(Continued)

BASES

ACTIONS
 (continued)

OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

C.2D.2

Reduction of THERMAL POWER to $\leq 50\%$ RTP puts the core into a condition where rod position is not significantly affecting core peaking factors limits (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1.1 D.1.1 and C.1.2 D.1.2 or reduce power to $\leq 50\%$ RTP.

D.1E.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
 REQUIREMENTS

SR 3.1.8.1

Verification that the DRPI agrees with the demand position within 12 steps ensures that the DRPI is operating correctly. Verification at 24, 48, 120, and 228 steps withdrawn for the control and shutdown banks provides assurance that the DRPI is operating correctly over the full range of indication. ~~Since the DRPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.~~

~~The 18 month frequency is based on the need to preform This surveillance under conditions that apply during a plant outage and the is performed prior to reactor criticality after each removal of the reactor head since there is potential for unnecessary plant transients if the SR were performed with the reactor at power. - Operating experience has shown these components usually pass the SR when performed at a Frequency of once every 18 months. Therefore, the frequency was concluded to be acceptable from a reliability standpoint.~~

(Continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
 2. FSAR, Chapter 15.
 3. ~~FSAR, Chapter 15, WCAP-10216-P/A, Rev. 1A, "Relaxation of Constant Axial Offset Control and E₀ Surveillance Technical Specification, February 1994.~~
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.10 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

- a. Ensure that the facility has been adequately designed;
- b. Validate the analytical models used in the design and analysis;
- c. Verify the assumptions used to predict unit response;
- d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and
- e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed (Ref. 4).

(Continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.10 PHYSICS TESTS Exceptions - MODE 2

BASES

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below: typically include:

- a. Critical Boron Concentration - Control Rods Withdrawn;
- b. Critical Boron Concentration - Control Rods Inserted;
- c. Control Rod Worth; and
- d. Isothermal Temperature Coefficient (ITC). and
- e. Neutron Flux Symmetry.

The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the

(Continued)

BASES

BACKGROUND
(continued)

- operating controls and process variables to deviate from their LCO requirements during their performance.
- a. ~~The Critical Boron Concentration Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical ($k_{eff} = 1.0$), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.~~
- b. ~~The Critical Boron Concentration Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.5, "Rod Group Alignment Limits"; LCO 3.1.6, "Shutdown Bank Insertion Limit"; or LCO 3.1.7, "Control Bank Insertion Limits."~~
- c. ~~The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core.~~

(Continued)

BASES

~~The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.5, LCO 3.1.6, or LCO 3.1.7.~~

- d- ~~The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of~~

(Continued)

BASES

BACKGROUND
 (continued)

~~performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."~~

- ~~e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at $\leq 30\%$ RTP (Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.5, LCO 3.1.6, or LCO 3.1.7. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at $\leq 30\%$ RT.~~

APPLICABLE
 SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

(Continued)

BASES

The FSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables [14.1.1 and 14.1.2] summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS 19.6.1-1985 (Ref. 4). Reload fuel cycle PHYSICS TESTS are performed in accordance with Technical Specification requirements, fuel vendor guidelines, and established industry practices. Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to $\leq 5\%$ RTP, the reactor coolant temperature is kept $\geq 531^\circ\text{F}$, and SDM is $\geq [1.6]\% \Delta k/k$ within the limits provided in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement 10 CFR 50.36(c)(2)(1).

BASES

Reference 6 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.4 ~~3~~, LCO 3.1.5 ~~4~~, LCO 3.1.6 ~~5~~, LCO 3.1.7 ~~6~~, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

- a. RCS lowest ~~operating~~ loop average temperature is \geq [531] °F; and
- b. SDM is \rightarrow [1.6] % $\Delta k/k$. ~~within the limits provided in the COLR.~~
~~and~~
- c. ~~THERMAL POWER IS \leq 5% RTP~~

APPLICABILITY

This LCO is applicable ~~in MODE 2~~ when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP. ~~Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, "PHYSICS TESTS Exceptions MODE 1."~~

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

(Continued)

BASES

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest operating loop's T_{avg} is < 531°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with an operating loop's temperature below 531°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

(Continued)

BASES

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE
 REQUIREMENTS

SR 3.1.108.1

The required power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each OPERABLE power range and intermediate range channels within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.108.2

Verification that the RCS lowest operating loop T_{avg} is $\geq 531^{\circ}\text{F}$ will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.108.3

Verification that the THERMAL POWER is $\geq 5\%$ RIP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 1 hour during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.108.4

Verification that the SDM is within limits specified in the COLR ensures that, for the specific RCCA and RCS temperature manipulations performed during PHYSICS TESTS, the plant is not operating in a condition that could invalidate the safety analysis assumptions.

(Continued)

BASES

~~The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:~~

- ~~a. RCS boron concentration;~~
- ~~b. Control bank position;~~
- ~~c. RCS average temperature;~~
- ~~d. Fuel burnup based on gross thermal energy generation;~~
- ~~e. Xenon concentration;~~
- ~~f. Samarium concentration; and~~
- ~~g. Isothermal temperature coefficient (ITC).~~

~~Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.~~

(Continued)

BASES

The SDM for physics testing during tests where traditional SDM monitoring techniques are not adequate is determined for the most restrictive test based on design calculations. Plant conditions are monitored during these tests to verify adequate SDM.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SD.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
 2. 10 CFR 50.59.
 3. Regulatory Guide 1.68, Revision 2, August, 1978.
 4. ~~ANSI/ANS 19.6.1 1985, December 13, 1985. Not used~~
 5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.
 6. WCAP-11618, including Addendum 1, April 1989.
-

Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(3 Pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.1

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER

JUSTIFICATION

- 3.1-1 In accordance with TSTF-9, Rev. 1, this change would relocate the specified limit for SDM from ITS to the COLR. This change occurs in several specifications including the specification for SDM and those specifications with ACTIONS that require verifying SDM within limits.
- 3.1-2 The Note for SR 3.1.2.1 indicates that predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 EFPD after each refueling. However, both the Bases for Specification 3.1.3 and the CTS requirements in Specification 3.1.1.5 state that the normalization shall be done prior to exceeding a fuel burnup of 60 EFPD after each refueling.
- 3.1-3 Not applicable to DCCP. See Conversion Comparison Table (Enclosure 6B).
- 3.1-4 SR 3.1.4.2 of NUREG-1431, Rev. 1 would be deleted. In accordance with TSTF-13, Rev. 1, the intent of this SR is only to determine the next frequency for SR 3.1.4.3. Performance of SR 3.1.4.2 is not necessary to assure that the LCO is met; SR 3.1.4.3 fulfills that purpose. Therefore, SR 3.1.4.2 may be deleted. In addition, the note in the frequency column of SR 3.1.4.2 would be moved to become Note 1 in the surveillance column of SR 3.1.4.3. This is for clarification purposes. As discussed in CN 3.1-9, section renumbering results in SR 3.1.4.3 of NUREG-1431, Rev. 1 becoming ITS SR 3.1.3.2.
- 3.1-5 Per CTS [3.1.3.1], the words "with all" have been removed from ITS LCO 3.1.4. This is a clarification that ensures the proper interpretation of the LCO. The change makes it clear that only one channel of DRPI is necessary to meet the alignment accuracy requirement of the LCO. With the word "all" in the statement, it may be possible for those unfamiliar with the DRPI design to interpret the LCO as applying to all channels of DRPI.
- 3.1-6 LCO 3.1.4 would be split into two separate statements to clarify that the alignment limit is separate from OPERABILITY of the control rod. The Condition A wording is broadened from "untrippable" to "inoperable" to ensure the Condition encompasses all causes of inoperability. Previous wording was ambiguous for rods that, for instance, had slow drop times but were still trippable. These slow rods are inoperable rods, and the change clarifies the appropriate ACTIONS. The Bases are changed to reflect the changes to the LCO and Condition A. These changes are based on TSTF-107.
- 3.1-7 This change to the ISTS would incorporate, into LCO 3.1.7, an ACTION statement that was previously approved as part of the Callaway and Wolf Creek licensing basis as revised in Enclosure 2. The ACTION statement would permit continued POWER OPERATION for up to 24 hours with more than one DRPI channel per rod group inoperable. The ACTION statement specifies additional Required ACTIONS beyond those applicable to the Condition of 1 DRPI channel per group inoperable. The Bases for this change also would be incorporated into the Bases for the plant ITS. These changes are consistent with Traveler WOG-73, Rev.1. The note under the ACTIONS is changed to be consistent with the new Required Actions.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.1

CHANGE
NUMBER

JUSTIFICATION

3.1-19

Not used.

3.1-20

Consistent with CTS 3/4.10.3, "Physics Tests," ITS LCO 3.1.8 and its Condition C and SR 3.1.8.2 are modified to refer to "operating" RCS loops. If an operating loop were secured while operating in MODE 2, Required Action A.1 of ITS 3.4.4 directs the plant to be in MODE 3 within 6 hours. However, depending on initial power level and plant Conditions, securing an RCS loop may result in a loop's average temperature of [531]°F fairly quickly due to the reverse flow through the secured loop and the cooling achieved by the steam generator. In this scenario, it would not be appropriate to invoke Required Action C.1 of ITS 3.1.8 with its 15 minute Completion Time to restore loop average temperature to within limits since the flow in the secured loop is not passing through the core and ITS LCO 3.4.4 provides appropriate corrective ACTION, consistent with CTS.

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(3 Pages)

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.1

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.1-1	In accordance with industry Traveler TSTF-9, Rev. 1, this change would relocate the specified limits for SDM from several TS to the COLR.	Yes	Yes	Yes	Yes
3.1-2	Changes the note to SR 3.1.2.1, which deals with verifying core reactivity within limits, to state that the normalization of predicted reactivity values to correspond to measured values shall be done prior to exceeding a fuel burnup of 60 EFPD after each refueling.	Yes	Yes	No, maintaining ITS wording.	No, maintaining ITS wording.
3.1-3	The Wolf Creek ITS LCO 3.1.6 Required Action C.1 is revised from "Be in MODE 3." to "Be in MODE 2 with $k_{eff} < 1.0$."	No	No	Yes	No
3.1-4	In accordance with industry Traveler TSTF-13, Rev. 1, SR 3.1.4.2, which requires verifying MTC within the 300 ppm boron limit, is deleted and the note in that SR is moved to the SR that requires the lower MTC limit to be verified. The deleted SR is not a requirement separate from the lower MTC verification SR, but is essentially a clarification of when the SR for the lower MTC limit should be performed.	Yes	Yes	Yes	Yes
3.1-5	Per CTS [3.1.3.1], the words "with all" are removed from the LCO for control rod alignment limits. This ensures that the number of channels of DRPI required to be OPERABLE will not be misconstrued.	Yes	Yes	Yes	Yes
3.1-6	In accordance with TSTF - 107, the change provides additional clarification that the alignment limits in the LCO are separate from the OPERABILITY of a control rod.	Yes	Yes	Yes	Yes
3.1-7	An ACTION statement that was previously approved as part of the current licensing basis of Callaway and Wolf Creek would be added to ITS 3.1.7, as revised in Enclosure 2. The ACTION statement would permit operation for up to 24 hours with more than one digital rod position indicator per group inoperable.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.1

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.1-8	In accordance with TSTF - 89, the requirement to compare DRPI against group demand position would be required whenever the reactor vessel head is removed, not every 18 months.	Yes	Yes	Yes	Yes
3.1-9	This change would eliminate ITS 3.1.2 because the SDM requirements for MODE 5 have been incorporated into Specification 3.1.1 in accordance with Traveler TSTF-136.	Yes	Yes	Yes	Yes
3.1-10	Several surveillances (e.g., rod position deviation monitor and rod insertion limit monitor in this section) contain ACTIONS in the form of increased surveillance frequency to be performed in the event of inoperable alarms. These ACTIONS are relocated from the TS to licensee controlled documents. This is consistent with TSTF-110, Rev 1.	Yes	Yes	Yes	Yes
3.1-11	Not Used.	N/A	N/A	N/A	N/A
3.1-12	The required ACTIONS for inoperable DRPI are revised per the current licensing basis to note that the use of movable incore detectors for rod position verification is an indirect assessment at best. The position of some rods can not be ascertained by this method.	Yes	Yes	Yes	Yes
3.1-13	In accordance with traveler TSTF-14, Rev. 3, the LCO and SR are modified to verify that THERMAL POWER \leq 5% RTP. This provides an LCO requirement to correspond to Condition B which requires RTP to be within limit.	Yes	Yes	Yes	Yes
3.1-14	Not used.	N/A	N/A	N/A	N/A
3.1-15	In accordance with TSTF-12, Rev. 1, this change would delete its LCOs 3.1.9 and 3.1-11. This change and TSTF-136 rennumbers ITS 3.1.10 to ITS 3.1.8.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.1

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.1-16	This change adds the requirement to perform SR 3.2.1.2 in addition to SR 3.2.1.1 during performance of ITS 3.1.4 Required Action B 2.4, consistent with traveler WOG-105.	Yes	Yes	Yes	Yes
3.1-17	Consistent with CTS LCO 3.1.3.2 and the wording of ITS 3.1.7, Condition A and B, ITS 3.1.7, Condition C is clarified to state that the inoperable position indicators are inoperable position indicators are inoperable DRIPs.	Yes	Yes	Yes	Yes
3.1-18	A MODE change restriction has been added to ITS 3.1.1, in LCO Applicability, per the matrix discussed in CN 01-02-LS1 of the 3.0 Package.	Yes	Yes	Yes	Yes
3.1-19	Not used.	N/A	N/A	N/A	N/A
3.1-20	Consistent with CTS 3/4.10.3, "Physics Tests," ITS LCO 3.1.8 and its Condition C and SR 3.1.8.2 are modified to refer to "operating" RCS loops.	Yes	Yes	Yes	Yes

JLS Conversion to Improved Technical Specifications Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 3/4.2 - Power Distribution Limits

ITS 3.2 - Power Distribution Limits



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.2

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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

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CROSS-REFERENCE TABLE FOR 3/4.2
Sorted by Current TS

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.2.1	LCO			3.2.3	LCO		
3.2.1	APP		01-12-A	3.2.3	APP		
3.2.1	APP	Note	01-01-A	Not used			
3.2.1	ACTION	a.1	01-13-A	Not used			
3.2.1	ACTION	a.2	01-16-LS9	3.2.3	CONDITION	A	
3.2.1	ACTION	b	01-12-A	Not used			
4.2.1.1	SR	a.1	01-07-LG	3.2.3.1	SR		
4.2.1.1	SR	a.2	01-07-LG	Not used			3.2-10
4.2.1.2	SR		01-09-A	3.2.3	LCO	NOTE	
3.2.2	LCO		02-01-LG	3.2.1	LCO		
3.2.2	APP			3.2.1	APP		
3.2.2	ACTION		02-01-LG	3.2.1	CONDITION	A	
3.2.2	ACTION	a	02-02-LS3	3.2.1	CONDITION	A.1 - A.3	3.2-06
3.2.2	ACTION	a	02-01-LG	3.2.1	CONDITION	A.3	
3.2.2	ACTION	a	02-11- LS15	Not used			
3.2.2	ACTION	b		3.2.1	CONDITION	A.4	
3.2.2	ACTION	NEW	02-03-M	3.2.1	CONDITION	C	
4.2.2.1	SR		02-07-A	3.2.1.1	SR	NOTE	3.2-13
4.2.2.2	SR	a, b, c	02-01-LG	Not used			
4.2.2.2	SR	NEW	02-01-LG	3.2.1.1 3.2.1.2	SR		
4.2.2.2	SR	d	02-12-A	3.2.1.1 3.2.1.2	SR		
4.2.2.2	SR	d Note	02-07-A	3.2.1.1 3.2.1.2	SR	NOTE	3.2-13
4.2.2.2	SR	d.1	02-06-A	3.2.1.1 3.2.1.2	SR		3.2-02 3.2-12
4.2.2.2	SR	d.2		3.2.1.1 3.2.1.2	SR		
4.2.2.2	SR	NEW	02-05-M	3.2.1.1 3.2.1.2	SR		

CROSS-REFERENCE TABLE FOR 3/4.2
Sorted by Current TS

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
4.2.2.2	SR	e		3.2.1.2	SR	NOTE	3.2-03 3.2-07
4.2.2.2	SR	f.1	02-01-LG	Not used			
4.2.2.2	SR	f.2.a	02-14-M	3.2.1	CONDITION	B	3.2-08
4.2.2.2	SR	f.2.a	02-03-M	3.2.1	CONDITION	C	
4.2.2.2	SR	f.2.b	02-09-M	Not used			
4.2.2.2	SR	g	02-01-LG	Not used			
4.2.2.3	SR		02-01-LG	Not used			
3.2.3	LCO		03-01-LG	3.2.2	LCO		
3.2.3	LCO		03-07-A	3.4.1.c	LCO		
3.2.3	LCO		03-08-A	3.4.1.c	LCO		3.4-41
3.2.3	LCO		03-01-LG	Not used			
3.2.3	APP			3.2.2. 3.4.1	APP		
3.2.3	ACTION		03-07-A	3.2.2. 3.4.1	CONDITION	A	
3.2.3	ACTION		03-08-A	3.4.1	CONDITION	A	
3.2.3	ACTION	a.1	03-09-M 03-07-A	3.4.1	CONDITION	A.1	
3.2.3	ACTION	a.2	03-02-LS5	3.2.2	CONDITION	A.1.1, A.1.2.1	
3.2.3	ACTION	a.2	03-07-A	3.2.2	CONDITION	A.1.1	
3.2.3	ACTION	a.2	02-02-LS3	3.2.2	CONDITION	A.1.2.2	3.2-06
3.2.3	ACTION	2. Note (NEW)	03-05-M	3.2.2	CONDITION	A, NOTE	
3.2.3	LCO	Fig. 3.2- 3a	03-08-A 3-10-LG	Not used			
3.2.3	LCO	Table (NEW)	03-08-A	3.4.1.c	LCO	Table 3.4.1-1	3.4-41
3.2.3	LCO	Fig. 3.2- 3b	03-08-A 3-10-LG	Not used			
3.2.3	LCO	Table (NEW)	03-08-A	3.4.1.c	LCO	Table 3.4.1-2	3.4.41
3.2.3	ACTION	b	03-07-A	3.2.2	CONDITION	A.2	

CROSS-REFERENCE TABLE FOR 3/4.2
Sorted by Current TS

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.2.3	ACTION	b	03-01-LG	Not used			
3.2.3	ACTION	b	03-04-LS6	3.2.2	CONDITION	B.1	
3.2.3	ACTION	c	03-05-M	3.2.2	CONDITION	A, NOTE	
3.2.3	ACTION	c	03-07-A	3.2.2	CONDITION	A.3	
3.2.3	ACTION	c	03-01-LG	Not used			
3.2.3	ACTION	c	03-09-M	3.4.1	CONDITION	A.1	
3.2.3	ACTION	NEW	03-03-M	3.2.2	CONDITION	B.1	
3.2.3	ACTION	c. Note (NEW)	03-06-A	3.2.2	CONDITION	A.3, NOTE	
4.2.3.1	SR		02-07-A	3.2.2.1	SR	NOTE	3.2-13
4.2.3.2	SR		03-07-A	3.2.2.1	SR		
4.2.3.2	SR		05-10-A	Not used			
4.2.3.3	SR		03-08-A	3.4.1.3	SR		3.4-41
4.2.3.3	SR		03-07-A	Not used			
4.2.3.4	SR		05-03-LG	3.3	SR		
4.2.3.5	SR			3.4.1.4	SR		3.4-40 3.4-38 3.4-41
3.2.4	LCO			3.2.4	LCO		
3.2.4	APP			3.2.4	APP		
3.2.4	APP	NOTE	01-01-A	Not used			
3.2.4	ACTION	a.	04-06- LS13	3.2.4	CONDITION	A	
3.2.4	ACTION	a.1	04-02- LS10	Not used			
3.2.4	ACTION	a.2	04-05- LS11	3.2.4	CONDITION	A.1	
3.2.4	ACTION	NEW	04-05- LS11	3.2.4	CONDITION	A.2	3.2-15
3.2.4	ACTION	NEW	04-05- LS11	3.2.4	CONDITION	A.3	3.2-16
3.2.4	ACTION	NEW	04-05- LS11	3.2.4	CONDITION	A.4	

CROSS-REFERENCE TABLE FOR 3/4.2
Sorted by Current TS

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.2.4	ACTION	NEW	04-01-A	3.2.4	CONDITION	A.5	3.2-05
3.2.4	ACTION	NEW	04-05-LS11	3.2.4	CONDITION	A.6	3.2-16
3.2.4	ACTION	NEW	04-05-LS11	3.2.4	CONDITION	B	
3.2.4	ACTION	a.3	04-05-LS11	Not used			
3.2.4	ACTION	a.4	04-05-LS11	Not used			
3.2.4	ACTION	b, c	04-06-LS13	Not used			
3.2.4	ACTION	d	04-07-A	Not used			
4.2.4.1	SR	Note (NEW)	04-09-A	3.2.4.1	SR	NOTES	3.2-15
4.2.4.1	SR	Note (NEW)	04-04-LS12	3.2.4.1	SR	NOTES	
4.2.4.1	SR	a	01-07-LG	3.2.4.1	SR		3.2-10
4.2.4.1	SR	b	01-07-LG	Not used			
4.2.4.2	SR		04-03-LG	3.2.4.2	SR		3.2-15
4.2.4.2	SR	Note (NEW)	04-04-LS12	3.2.4.2	SR	NOTE	3.2-15
3.2.5	LCO		05-01-LG	3.4.1.a 3.4.1.b	LCO		
3.2.5	LCO	Table	05-01-LG	3.4.1	LCO		
3.2.5	APP			3.4.1	APP		
3.2.5, 3.2.3	ACTION		05-06-LS8	3.4.1	CONDITION	A, B	
4.2.5.1	SR			3.4.1.1 3.4.1.2	SR		

CROSS-REFERENCE TABLE FOR 3/4.2
Sorted by Improved TS

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.2.2	LCO		02-01-LG	3.2.1	LCO		
3.2.2	APP			3.2.1	APP		
3.2.2	ACTION		02-01-LG	3.2.1	CONDITION	A	
3.2.2	ACTION	a	02-02-LS3	3.2.1	CONDITION	A.1 - A.3	3.2-06
3.2.2	ACTION	a	02-01-LG	3.2.1	CONDITION	A.3	
3.2.2	ACTION	b		3.2.1	CONDITION	A.4	
4.2.2.2	SR	f.2.a	02-14-M	3.2.1	CONDITION	B	3.2-08
4.2.2.2	SR	f.2.a	02-03-M	3.2.1	CONDITION	C	
3.2.2	ACTION	NEW	02-03-M	3.2.1	CONDITION	C	
4.2.2.1	SR		02-07-A	3.2.1.1	SR	NOTE	3.2-13
4.2.2.2	SR	d Note	02-07-A	3.2.1.1 3.2.1.2	SR	NOTE	3.2-13
4.2.2.2	SR	NEW	02-01-LG	3.2.1.1 3.2.1.2	SR		
4.2.2.2	SR	d	02-12-A	3.2.1.1 3.2.1.2	SR		
4.2.2.2	SR	d.1	02-06-A	3.2.1.1 3.2.1.2	SR		3.2-02 3.2-12
4.2.2.2	SR	d.2		3.2.1.1 3.2.1.2	SR		
4.2.2.2	SR	NEW	02-05-M	3.2.1.1 3.2.1.2	SR		
4.2.2.2	SR	e		3.2.1.2	SR	NOTE	3.2-03 3.2-07
3.2.3	LCO		03-01-LG	3.2.2	LCO		
3.2.3	APP			3.2.2, 3.4.1	APP		
3.2.3	ACTION	a.2	03-02-LS5	3.2.2	CONDITION	A.1.1, A.1.2.1	
3.2.3	ACTION	a.2	03-07-A	3.2.2	CONDITION	A.1.1	
3.2.3	ACTION	a.2	02-02-LS3	3.2.2	CONDITION	A.1.2.2	3.2-06
3.2.3	ACTION	2. Note (NEW)	03-05-M	3.2.2	CONDITION	A, NOTE	

CROSS-REFERENCE TABLE FOR 3/4.2
Sorted by Improved TS

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.2.3	ACTION	c	03-05-M	3.2.2	CONDITION	A, NOTE	
3.2.3	ACTION	b	03-07-A	3.2.2	CONDITION	A.2	
3.2.3	ACTION	c	03-07-A	3.2.2	CONDITION	A.3	
3.2.3	ACTION	c. Note (NEW)	03-06-A	3.2.2	CONDITION	A.3, NOTE	
3.2.3	ACTION	b	03-04-LS6	3.2.2	CONDITION	B.1	
3.2.3	ACTION	NEW	03-03-M	3.2.2	CONDITION	B.1	
4.2.3.1	SR		02-07-A	3.2.2.1	SR	NOTE	3.2-13
4.2.3.2	SR		03-07-A	3.2.2.1	SR		
3.2.1	LCO			3.2.3	LCO		
4.2.1.2	SR		01-09-A	3.2.3	LCO	NOTE	
3.2.1	APP		01-12-A	3.2.3	APP		
3.2.1	ACTION	a.2	01-16-LS9	3.2.3	CONDITION	A	
4.2.1.1	SR	a.1	01-07-LG	3.2.3.1	SR		
3.2.4	LCO			3.2.4	LCO		
3.2.4	APP			3.2.4	APP		
3.2.4	ACTION	a.	04-06- LS13	3.2.4	CONDITION	A	
3.2.4	ACTION	a.2	04-05- LS11	3.2.4	CONDITION	A.1	
3.2.4	ACTION	NEW	04-05- LS11	3.2.4	CONDITION	A.2	3.2-15
3.2.4	ACTION	NEW	04-05- LS11	3.2.4	CONDITION	A.3	3.2-16
3.2.4	ACTION	NEW	04-05- LS11	3.2.4	CONDITION	A.4	
3.2.4	ACTION	NEW	04-01-A	3.2.4	CONDITION	A.5	3.2-05
3.2.4	ACTION	NEW	04-05- LS11	3.2.4	CONDITION	A.6	3.2-16
3.2.4	ACTION	NEW	04-05- LS11	3.2.4	CONDITION	B	
4.2.4.1	SR	Note (NEW)	04-09-A	3.2.4.1	SR	NOTES	3.2-15

CROSS-REFERENCE TABLE FOR 3/4.2
Sorted by Improved TS

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
4.2.4.1	SR	Note (NEW)	04-04-LS12	3.2.4.1	SR	NOTES	
4.2.4.1	SR	a	01-07-LG	3.2.4.1	SR		3.2-10
4.2.4.2	SR		04-03-LG	3.2.4.2	SR		3.2-15
4.2.4.2	SR	Note (NEW)	04-04-LS12	3.2.4.2	SR	NOTE	3.2-15
4.2.3.4	SR		05-03-A	3.3	SR		
3.2.3	LCO		03-07-A	3.4.1.c	LCO		
3.2.3	LCO		03-08-A	3.4.1.c	LCO		3.4-41
3.2.3	LCO	Table (NEW)	03-08-A	3.4.1.c	LCO	Table 3.4.1-1	3.4-41
3.2.3	LCO	Table (NEW)	03-08-A	3.4.1.c	LCO	Table 3.4.1-2	3.4.41
3.2.5	LCO		05-01-LG	3.4.1.a 3.4.1.b	LCO		
3.2.5, 3.2.3	APP			3.4.1	APP		
3.2.5	LCO	Table	05-01-LG	3.4.1	LCO	A	
3.2.3	ACTION		03-08-A	3.4.1, 3.2.2	CONDITION		
3.2.3	ACTION		03-07-A	3.4.1	CONDITION	A	
3.2.3	ACTION	a.1	03-07-A 03-09-M	3.4.1	CONDITION	A.1	
3.2.3	ACTION	c	03-09-M	3.4.1	CONDITION	A.1	
3.2.5	ACTION		05-06-LS8	3.4.1	CONDITION	A, B	
4.2.5.1	SR			3.4.1.1 3.4.1.2	SR		
4.2.3.3	SR		03-08-A	3.4.1.3	SR		3.4-41
4.2.3.5	SR			3.4.1.4	SR		3.4-40 3.4-38 3.4-41

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

**Methodology for Cross-Reference Tables
(Continued)**

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).
Note:	The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
3.2.1	3/4 2-1
3.2.2	3/4 2-5
3.2.3	3/4 2-13
3.2.4	3/4 2-18
3.2.5	3/4 2-21

Methodology (2 Pages)

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITIONS FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODE 1 ABOVE ~~GREATER THAN OR EQUAL TO~~ 50 PERCENT RATED THERMAL POWER*. 01-12-A

ACTION:

- a. With the indicated AXIAL FLUX DIFFERENCE outside of the limits specified in the COLR.
- ~~1. Either restore the indicated AFD to within the limits within 15 minutes, or~~ 01-13-A
 2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours. 01-16-LS9
- ~~b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.~~ 01-12-A

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 50 percent of RATED THERMAL POWER by:

- a. Monitoring the indicated AFD for each OPERABLE excore channel:
1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and 01-07-LG
 - ~~2. At least once per hour when the AFD Monitor Alarm is inoperable.~~ 01-07-LG

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least 2 OPERABLE excore channels are indicating the AFD to be outside the limits. 01-09-A

~~*See Special Test Exceptions Specification 3.10.2~~ 01-01-A

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POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-F₀(Z)

LIMITING CONDITION FOR OPERATION

3.2.2 F₀(Z) as approximated by F₀^R(Z) and F₀^H(Z) shall be within the limits specified in the COLR shall be limited by the following relationships:

02-01-LG

$$\frac{F(Z)}{Q} < \frac{[F^{RTP}]}{Q} [K(Z)] \text{ for } P > 0.5$$

$$\frac{F(Z)}{Q} < \frac{[F^{RTP}]}{0.5Q} [K(Z)] \text{ for } P \leq 0.5$$

where $\frac{F^{RTP}}{Q}$ = the F limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

and $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

$K(Z) =$ the normalized F₀(Z) for a given core height specified in the COLR.

APPLICABILITY: MODE 1

ACTION:

With ~~F₀^R(Z)~~ E₀(Z) exceeding its limit:

02-01-LG

- a. Reduce THERMAL POWER at least 1% for each 1% ~~F₀^R(Z)~~ E₀(Z) exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 72 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% ~~F₀^R(Z)~~ E₀(Z) exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.
 - 02-02-LS3
 - 02-01-LG
 - 02-11-LS15
- b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided F₀(Z) is demonstrated through incore mapping to be within its limit.

(new) If the above ACTIONS are not completed within required completion time, be in MODE 2 within the next 6 hours.

02-03-M

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POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

~~4.2.2.1 The provisions of Specification 4.0.4 are not applicable.~~

02-07-A

4.2.2.2 $F_0(Z)$ shall be evaluated to determine if ~~it~~ $E_0(Z)$ is within its limits by:

~~(new) Verifying $F_0^C(Z)$ and $F_0^M(Z)$ satisfy the relationships in the COLR~~

02-01-LG

~~a. Using the moveable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.~~

~~b. Increasing the measured $F_0(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.~~

02-01-LG

~~c. Satisfying the following relationship:~~

~~$$F_0^M(Z) \leq \frac{E_0^{RTP}}{P} \times K(Z)$$
 for $P > 0.5$
$$P \times W(Z)$$~~

~~$$F_0^M(Z) \leq \frac{E_0^{RTP}}{W(z) \times 0.5} \times K(Z)$$
 for $P \leq 0.5$~~

~~where $F_0^M(Z)$ is the measured $F_0(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, E_0^{RTP} is the F_0 limit, $K(z)$ is the normalized $F_0(z)$ as a function of core height, P is the relative THERMAL POWER, and $W(z)$ is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_0^{RTP} , $K(z)$, and $W(z)$ are specified in the COLR.~~

02-07-A

~~d. Measuring $F_0^M(z)$, $F_0^C(Z)$ and $F_0^N(Z)$ according to the following schedule:~~

02-12-A

~~(new) Once after each refueling prior to THERMAL POWER exceeding 75% RTP~~

02-05-M

~~1. Once within 12 24 hours after Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_0^C(Z)$ $E_0(z)$ was last determined.* or and~~

02-06-A

~~2. At least once per 31 Effective Full Power Days (EFPD) thereafter, whichever occurs first.~~

~~e. With measurements indicating~~

~~maximum
$$\frac{F_0^M(Z)}{K(Z)}$$
 over z~~

~~has increased since the previous determination of $F_0^M(Z)$ either of the following actions shall be taken:~~

~~*During power escalation at the beginning of each cycle following shutdown, power level may be increased until a an equilibrium power level for extended operation has been achieved and a power distribution map obtained.~~

02-07-A

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

- 1) $F_0^M(Z)$ shall be increased over that specified in Specification 4.2.2.2.c by an appropriate factor specified in the COLR, or 02-01-LG
- 2) $F_0^M(Z)$ shall be measured at least once per 7 EFPD until two successive maps indicate that
- maximum $F_0^{CM}(Z)$ is not increasing.
over Z $K(Z)$

~~f. With the relationship specified in Specification 4.2.2.2.c above not being satisfied:~~

- ~~1) Calculate the percent $F_0(Z)$ if $F_0^M(Z)$ exceeds its limit by the following expression:~~

$$\left\{ \left(\frac{\text{maximum over } z \left[\frac{F_0^M(z) \times W(z)}{F_0^{RIP}} \right] - 1}{p} \times K(z) \right) \right\} \times 100 \text{ for } p \geq 0.5$$

$$\left\{ \left(\frac{\text{maximum over } z \left[\frac{F_0^M(z) \times W(z)}{F_0^{RIP}} \right] - 1}{0.5} \times K(z) \right) \right\} \times 100 \text{ for } p < 0.5$$

2. ~~Either one of the following actions shall be taken:~~ 02-14-M
- a) ~~Place the core in an equilibrium condition where the limit in Specification 4.2.2.2.c is satisfied. Power level may then be increased provided the Within 4 hours, reduce AFD limits of Specification 3.2.1 are reduced at least 1% AFD for each percent $F_0^M(Z)$ exceeds its limit, or be in Mode 2 within the next 6 hours.~~ 02-03-M

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

~~b) Comply with the requirements of Specification 3.2.2 for $F_0(Z)$ exceeding its limit by the percent calculated.~~

02-09-M

~~g. The limits specified in Specification 4.2.2.2.c, 4.2.2.2.e, and 4.2.2.2.f above are not applicable in the following core plane regions:~~

02-01-LG

- ~~1. Lower core region from 0 to 15%, inclusive.~~
- ~~2. Upper core region from 85 to 100%, inclusive.~~

~~4.2.2.3 When $F_0(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_0(Z)$ shall be obtained from power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.~~

02-01-LG

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POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta U}^N$ shall be within the limits specified in the COLR. The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Table 3.2-1 Figure 3.2-3a for Unit 1 and Table 3.2-2 Figure 3.2-3b for Unit 2 for four loop operation.

03-01-LG

03-07-A

Where:

03-08-A

a. $R = \frac{F_{\Delta U}^N}{F_{\Delta U}^{RTP} [1.0 + PF_{\Delta U} (1.0 - P)]}$ for LOPAR fuel

$R = \frac{F_{\Delta U}^N}{F_{\Delta U}^{RTP} [1.0 + PF_{\Delta U} (1.0 - P)]}$ for VANTAGE 5 fuel

03-01-LG

b. $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

c. $F_{\Delta U}^N$ = Measured values of $F_{\Delta U}^N$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F_{\Delta U}^N$ shall be used to calculate R since Figure 3.2-3a for Unit 1 and Figure 3.2-3b for Unit 2 include measurement uncertainties of 2.4% for flow and 4% for incore measurement of $F_{\Delta U}^N$.

d. $F_{\Delta U}^{RTP}$ = The $F_{\Delta U}^N$ limits at Rated Thermal Power (RTP) specified in the Core Operating Limits Report (COLR).

e. $PF_{\Delta U}$ = The Power Factor Multipliers specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With $F_{\Delta U}^N$ not within its limit, or with the combination of RCS total flow rate and R outside the region of acceptable operation shown on Table 3.2-1 Figure 3.2-3a for Unit 1 and Table 3.2-2 Figure 3.2-3b for Unit 2:

03-07-A

03-08-A

a. Within 2 hours either:

03-09-M

1. Restore the combination of RCS total flow rate and R to within the above limits, or

03-07-A

2. Within 4 hours either:

03-02-LS5

Restore $F_{\Delta U}^N$ to within limits, or

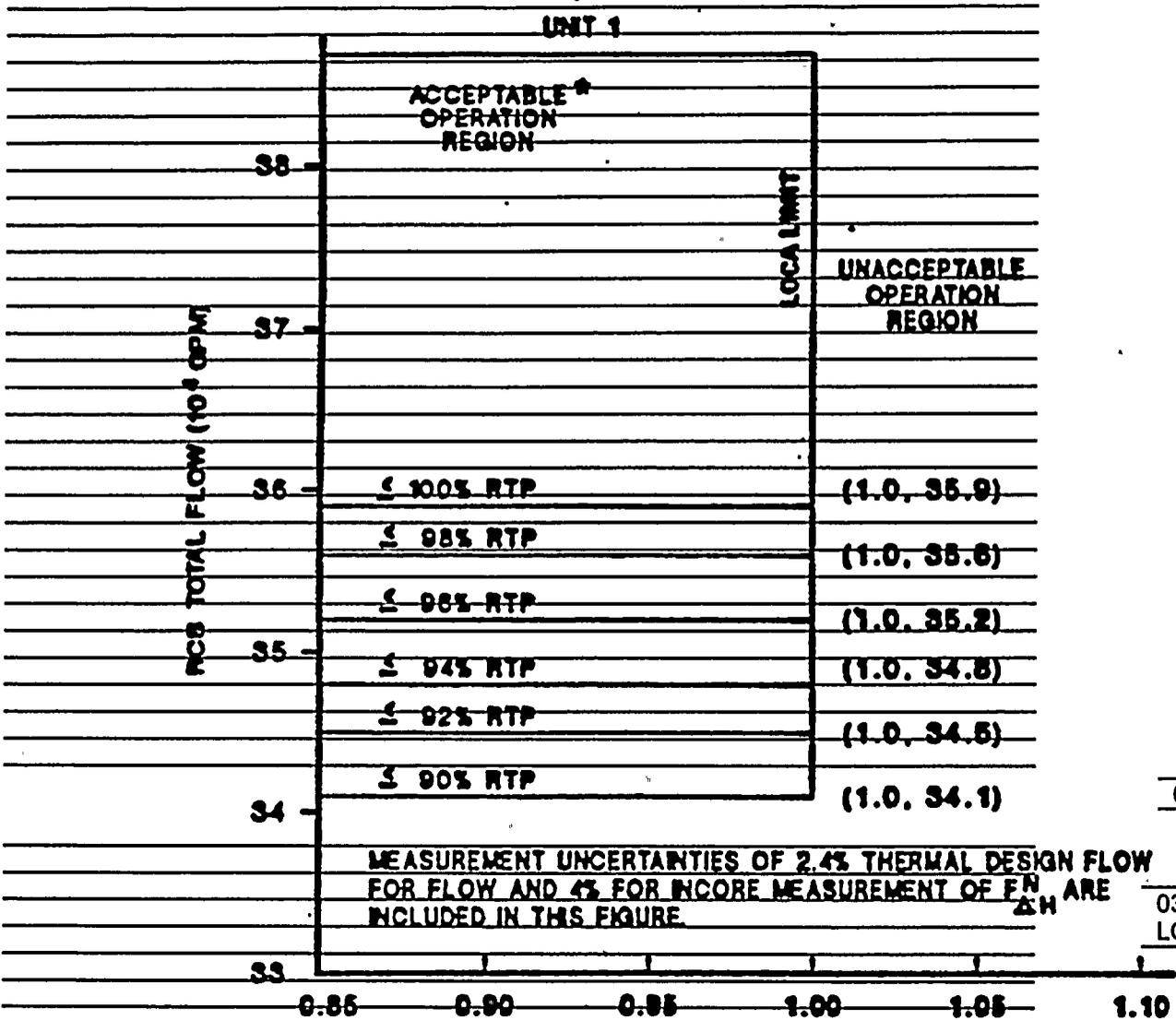
03-07-A

Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 68 4-hours.

02-02-LS3

(new)* Actions b. and c. must be completed whenever $F_{\Delta U}^N$ exceeds its limit

03-05-M



03-08-A

03-10-LG

$$R = \frac{F_{\Delta H}^N}{F_{\Delta H}^{RTP}} (1.0 + PF_{\Delta H} (1.0 - P))$$

LOPAR FUEL

$$R = \frac{F_{\Delta H}^N}{F_{\Delta H}^{RTP}} (1.0 + PF_{\Delta H} (1.0 - P))$$

VANTAGE 5 FUEL

*WHEN OPERATING IN THE RESTRICTED POWER REGION, THE RESTRICTED POWER LEVEL SHALL BE CONSIDERED 100% RTP FOR FIGURE 2.1-1

03-08-A

FIGURE 3.2 3a
RCS TOTAL FLOWRATE VERSUS R(UNIT 1)

NEW (page 1 of 1)
Reduction in Percent Rated Thermal Power for Reduced RCS Flow Rate
Unit 1

03-08-A

Table 3.2-1

RCS Total Flow ^(a) (10 ⁴ GPM)	Acceptable Operating Region ^(b) (% RTP)
≥ 35.9	≤ 100%
≥ 35.6	≤ 98%
≥ 35.2	≤ 96%
≥ 34.8	≤ 94%
≥ 34.5	≤ 92%
≥ 34.1	≤ 90%

(a) For RCS Total Flow < 341 000 GPM, entry into Action a is required.

(b) When operating with restricted power limits, the restricted power level shall be considered 100% RTP for Figure 2.1.1-1.

NEW (page 1 of 1)
Reduction in Percent Rated Thermal Power for Reduced RCS Flow Rate
Unit 2

03-08-A

Table 3.2-2

RCS Total Flow ^(a) (10 ⁶ GPM)	Acceptable Operating Region ^(b) (% RTP)
≥ 36.3	≤ 100%
≥ 35.9	≤ 98%
≥ 35.5	≤ 96%
≥ 35.2	≤ 94%
≥ 34.8	≤ 92%
≥ 34.4	≤ 90%

(a) For RCS Total Flow < 344,000 GPM, entry into Action a. is required.

(b) When operating with restricted power limits, the restricted power level shall be considered 100% RTP for Figure 2.1.1-1.

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POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

- b. Within 24 hours of initially being outside the above limits, verify that F_{RH} is within limits through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 2 hours. 03-07-A
03-01-LG
03-04-LS6
- c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2 and/or b. above; subsequent POWER OPERATION may proceed provided that F_{RH} the combination of R and indicated RCS total flow rate are as demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within limits the region of acceptable operation shown on Figure 3.2 3a for Unit 1 and Figure 3.2 3b for Unit 2 prior to exceeding the following THERMAL POWER levels: 03-05-M
03-07-A
03-01-LG
1. A nominal 50% of RATED THERMAL POWER, 03-06-A
 2. A nominal 75% of RATED THERMAL POWER, and
 3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

~~(new) With the above required Actions and Completion Times not met, be in Mode 2 within the next 6 hours.~~ 03-03-M

SURVEILLANCE REQUIREMENTS

- 4.2.3.1 The provisions of Specification 4.0.4 are not applicable.
- 4.2.3.2 F_{RH} The combination of indicated RCS total flow rate and R shall be determined to be within limits the region of acceptable operation of Figure 3.2 3a for Unit 1 and Figure 3.2 3b for Unit 2. 02-07-A
03-07-A
- a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and 05-10-A
 - b. At least once per 31 Effective Full Power Days.
- 4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of ~~Table 3.2 1~~ Figure 3.2 3a for Unit 1 and ~~Table 3.2 2~~ Figure 3.2 3b for Unit 2 at least once per 12 hours when the value of R, obtained per Specification 4.2.3.2, is assumed to exist. 03-08-A
03-07-A
- 4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. 05-03-LG
- 4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

~~# THERMAL POWER does not have to be reduced to comply with this Action.~~ 03-06-A

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*.

ACTION:

- a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09: 04-06-LS13
 - 1. ~~Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:~~ 04-02-LS10
 - a) ~~The QUADRANT POWER TILT RATIO is reduced to within its limit, or~~
 - b) ~~THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.~~
 - 2. Within 2 hours either:
 - a) ~~Reduce the QUADRANT POWER TILT RATIO to within its limit, or~~
 - b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux High Trip Setpoints within the next 4 hours. 04-05-LS11
04-06-LS13
- ~~(new *) At least once per 12 hours, calculate QUADRANT POWER TILT RATIO and reduce THERMAL POWER by at least 3% from RATED THERMAL POWER for each 1% of QUADRANT POWER TILT RATIO in excess of 1, and~~ 04-05-LS11
04-06-LS13
- ~~(new) Within 24 hours, and once per 7 days thereafter, confirm that the Heat Flux Hot Channel Factor $F_{CH}(Z)$ is within its limit by performing Surveillance Requirement 4.2.2.2 and confirm that Nuclear Enthalpy Rise Hot Channel Factor F_{NH} is within its limit by performing Surveillance Requirement 4.2.3.2.~~ 04-05-LS11
- ~~(new **) Prior to increasing THERMAL POWER above the limit of Actions a.2.b and new*~~ 04-05-LS11
- ~~Re-evaluate the safety analyses and confirm that the results remain valid for the duration of operation under this condition, and then~~ 04-06-LS13
- ~~Normalize excore detectors to eliminate tilt.~~ 04-01-A
- ~~After Action (new**) is implemented and within 24 hours after reaching RTP or within 48 hours after increasing THERMAL POWER above the limit of Required Actions a.2.b and (new*), confirm that $F_{CH}(Z)$ is within its limits by performing Surveillance Requirement 4.2.2.2 and that F_{NH} is within its limits by performing Surveillance Requirement 4.2.3.2, and~~ 04-05-LS11
- ~~If the requirements above are not met, reduce thermal power to \leq 50% RATED THERMAL POWER within the next 4 hours.~~ 04-05-LS11
- 3. ~~Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and~~ 04-06-LS13
04-05-LS11

~~4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~

04-05-LS11

~~b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:~~

04-06-LS13

~~1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:~~

~~*See Special Test Exceptions Specification 3.10.2~~

01-01-A

POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

- a) ~~The QUADRANT POWER TILT RATIO is reduced to within its limit, or~~
- b) ~~THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.~~
- 2- ~~Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;~~
- 3- ~~Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and~~
- 4- ~~Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.~~
- e- ~~With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod, calculate the QUADRANT POWER TILT RATIO at least once per hour until THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER and:~~ 04-06-LS13
- 1- ~~Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours, and~~
- 2- ~~Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.~~
- d- ~~The provisions of Specification 3.0.4 are not applicable to POWER OPERATION above 50% of RATED THERMAL POWER.~~ 04-07-A

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:*

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and 01-07-LG

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable. 01-07-LG

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from the 4 pairs of symmetric thimble locations or from a full incore flux map per specification 3.3.3.2, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.** 04-03-LG

(new)* With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER \geq 75% RTP, the remaining three power range channels can be used for calculating QPTR. SR 4.2.4.2 may be performed in lieu of this surveillance. 04-09-A
04-04-LS12

(new)** Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER \geq 75% RTP. 04-04-LS12

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

- a. Reactor Coolant System T_{avg} , and
- b. Pressurizer Pressure.

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

05-06-LS8

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER

LIMITS

Actual Reactor Coolant System T_{avg}

$\leq 584.3^{\circ}\text{F}$

Actual Pressurizer Pressure

$\geq 2212 \text{ psia}^*$

05-01-LG

* Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. **Deletions** - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. **Additions** - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. **Modifications** - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. **Administrative** - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions** - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions** - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications** - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative** - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

Methodology For Mark-Up of Current TS
(Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers (1 Page)

Description of Changes (7 Pages)

TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS

SECTION 3/4.2

Technical Specification Title	CHG NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Axial Flux Difference (AFD)	01	3.2.1 (RAOC)	3.2.1 (RAOC)	3.2.1 (CAOC)	3.2.1 (RAOC)
Heat Flux Hot Channel Factor - $F_Q(Z)$	02	N/A	N/A	3.2.2	3.2.2
Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^N$	03	3.2.3	N/A	3.2.3	N/A
Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}(X,Y)$	03	NA	3.2.3	N/A	N/A
RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor	03	N/A	N/A	N/A	3.2.3
Quadrant Power Tilt Ratio	04	3.2.4	3.2.4	3.2.4	3.2.4
DNB Parameters	05	3.2.5	3.2.5	3.2.5	3.2.5

DESCRIPTION OF CHANGES TO TS SECTION 3/4.2

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	A	Removes reference to the special test exception in accordance with the format of NUREG-1431. This modification is acceptable because it is an administrative change which does not change the requirement only the format.
01-02	LS1	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-03	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-04	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-05	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-06	LS2	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-07	LG	As described in industry Traveler TSTF-110 which modifies NUREG-1431 requirements, several surveillances contain ACTIONS (in the form of increased surveillance frequency) to be performed in the event of inoperable alarms. These ACTIONS are moved to licensee-controlled documents. The alarms themselves do not directly relate to the limiting condition for operation (LCO) limits. This detail is not required to be in the TS to provide adequate protection of the public health and safety. Therefore, moving this detail is acceptable.
01-08		Not used.
01-09	A	To be consistent with NUREG 1431, the description of when the axial flux difference (AFD) is considered to be outside its [limits] will be a note to the ITS LCO, instead of the CTS surveillance requirement (SR). The technical content is equivalent. Therefore, this change is acceptable.
01-10	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-11	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-12	A	Applicability of the CTS is changed from greater than 50 percent rated thermal power (RTP) to greater than or equal to 50 percent RTP. The requirement to meet AFD limits prior to increasing power above 50 percent would be deleted. The applicability change is consistent with the AFD allowed operational space in the CORE OPERATING LIMITS REPORT (COLR), which is defined for 50 percent RTP and above. The elimination of the requirement to meet AFD requirements would have no impact on plant operations because it is understood that it is necessary to comply with improved technical specification (ITS) requirements.

**CHANGE
NUMBER****NSHC****DESCRIPTION**

01-13	A	The ACTION statement regarding restoring AFD to within limits within 15 minutes would be deleted. This has no effect on the time allowed for completion of required actions and restoring AFD to within limits is implicit in requirements for exiting the ACTION statement. Therefore, this change is administrative.
01-14		Not used.
01-15	M	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-16	LS9	Eliminates the requirement to reduce the power range neutron flux - high reactor trip setpoints when AFD is outside the RELAXED AXIAL OFFSET CONTROL (RAOC) limits.
02-01	LG	In accordance with NUREG-1431, the proposed change moves the details of the entire CTS LCO of the allowable $F_Q(Z)$ values, including the $K(Z)$ and $W(Z)$ parameters to the CORE OPERATING LIMITS REPORT (COLR) and/or Bases. Previously, only the full power value of $F_Q(Z)$, in addition to $K(Z)$ and $W(Z)$ had been in the COLR. Now, the dependence of $F_Q(Z)$ on THERMAL POWER is also located in the COLR. Details of the $F_Q(Z)$ measurement, including the treatment of uncertainties, are moved to the Bases. The Required Actions are rewritten for consistency with NUREG-1431. The specific changes include the more appropriate use of $F_Q^C(Z)$ and $F_Q^W(Z)$ versus $F_Q(Z)$.
02-02	LS3	The Required Actions are rewritten for consistency with NUREG-1431 and industry Traveler TSTF-95. The specific changes include the relaxation of the Completion Time requirement to reduce the high neutron flux reactor Trip Setpoints [from 4 hours] to 72 hours. The reduction of the setpoints is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the amount of work required to be done to reduce the setpoints, the small likelihood of a severe transient in this time period, and the prompt reduction in THERMAL POWER required upon discovery of the out-of-limit condition.
02-03	M	For consistency with NUREG-1431, the requirement is added to be in at least MODE 2 within 6 hours should any of the ACTIONS not be completed within the required time. This requirement is more restrictive than the previous requirement to enter CTS 3.0.3, which allowed 1 hour before the 6 hour shutdown requirement became effective.
02-04	M	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-05	M	Consistent with NUREG-1431, $F_Q^C(Z)$ and $F_Q^W(Z)$ must be verified to be within limits prior to exceeding 75 percent RTP after each refueling. This requirement is not explicit in the CTS. The TS are made more restrictive by stating this requirement.
02-06	A	Consistent with the Bases of CTS 4.0.3, which allows 24 hours for completing SRs that become applicable when an exception to Specification 4.0.4 is allowed, the frequency for assessing $F_Q(Z)$ is clarified by requiring that the measurement be performed within 24 hours after reaching equilibrium Conditions.

**CHANGE
NUMBER****NSHC****DESCRIPTION**

02-07	A	The footnote allowing the power to be increased until the THERMAL POWER for extended operation has been achieved has been incorporated in the note preceding SR 3.2.1.1 and SR 3.2.2.1 in the ITS allowing power to be increased until an equilibrium power level has been achieved. This footnote replaces the Specification 4.0.4 exemption in the CTS. Therefore, the change is administrative, and no technical changes would result.
02-08	LS4	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-09	M	The optional action to comply with CTS 3.2.2, if $F_o^W(Z)$ exceeds its limit would be deleted. This eliminates an option and is more restrictive.
02-10		Not used.
02-11	LS15	The proposed change, consistent with NUREG-1431, would delete the requirement that the reactor be in at least HOT STANDBY while performing the overpower ΔT Trip Setpoint reduction. It is sufficient to reduce power 1 percent for each 1 percent $F_o(z)$ exceeds its limit and then perform the required trip setpoint reduction at reduced power. The CTS requirement to be in at least HOT STANDBY is not a Westinghouse design basis requirement.
02-12	A	$F_o^W(Z)$ must be verified to be within limits whenever $F_o(Z)$ is measured as required by the CTS.
02-13	LG	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-14	M	This change is similar to CN 02-08-LS4 for the JLS. For DCP, there is currently no time limit; therefore, the imposition of a time limit is a more restrictive change.
03-01	LG	The details of the $F_{\Delta H}^N$ limits would be moved to the COLR. Previously, the equation for the dependence of $F_{\Delta H}^N$ on THERMAL POWER had been located in the LCO and the COLR. The full power limit value of $F_{\Delta H}^N$ and the power factor multiplier had been located only in the COLR. Now, the equation is also located only in the COLR. Definitions and details of the measurement, including the treatment of uncertainties, are moved to the Bases. The Required Actions are rewritten for consistency with NUREG-1431. The changes are acceptable because they remove details not required to be in TS to support operational safety.
03-02	LS5	The Completion Times would be revised to be consistent with NUREG-1431. The adequacy of these completion times is discussed in the applicable Bases section of NUREG-1431. In summary, 4 hours (versus 2 hours in the CTS) is provided to attempt to restore $F_{\Delta H}^N$ to within its limit or to reduce power to below 50 percent RTP.
03-03	M	The requirement to reduce power to less than or equal to 5 percent RTP (exit MODE 1) within the next 6 hours is added in lieu of the use of CTS LCO 3.0.3. This requirement is more restrictive than the previous requirement to enter LCO 3.0.3, because LCO 3.0.3 allowed 1 hour before the 6-hour shutdown requirement became effective.
03-04	LS6	With the Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$) having been outside its limits, the current TS require that within 24 hours after exceeding the $F_{\Delta H}^N$ limit, an incore flux map be performed to verify that the $F_{\Delta H}^N$ has been restored

CHANGE NUMBER

NSHC

DESCRIPTION

to within its limits. If this activity is not completed, the CTS require that the plant be taken to MODE 2 within the next 2 hours. The proposed change, revises the time allowed to reduce the reactor power to a Condition where the LCO does not apply (MODE 2, < 5 percent RTP) to be consistent with NUREG-1431. The adequacy of these Completion Times are discussed in the applicable Bases section of NUREG-1431. In summary, 6 hours (versus 2 hours in the CTS) is provided to perform an orderly shutdown of the plant. The proposed change is acceptable based on operational experience regarding performance of an orderly plant shutdown together with the negligible probability of an accident occurring during the extended shutdown interval.

03-05

M

If the enthalpy rise hot channel factor ACTION statements requiring flux mapping and correction of the cause or power reductions are entered, they must be completed, even if compliance with the LCO is restored. These requirements from NUREG-1431 are more restrictive than the corresponding requirements from CTS.

03-06

A

Consistent with NUREG-1431, a note would be added to state that THERMAL POWER does not need to be reduced below the power required by ACTION A. in order to comply with the series of flux maps required by ACTION C. This is a clarification of the CTS in that if compliance with the LCO is restored prior to reducing power level below 50 percent, flux maps need only be performed for those plateaus traversed. If power level did not drop below 95 percent, no flux map would be required. No flux map would be required by ACTION C, but would be required by ACTION [B].

03-07

A

The Parameter R, which is a derived value based on $F_{\Delta H}^N$, would be deleted. Using a combination of reactor coolant system (RCS) total flow rate and R to determine operation in the acceptable region of CTS Figures 3.2-3a and 3.2-3b would be deleted. RCS total flow would be moved to the departure from nucleate boiling (DNB) parameters specification. $F_{\Delta H}^N$ would be a separate specification from the requirements of RCS flow.

This is an administrative change since the limit for R in Figures 3.2-3a and 3.2-3b is a constant value (1.00); therefore, the equation deriving the Parameter R can be reduced and shown in terms of $F_{\Delta H}^N$.

03-08

A

Figures 3.2-3a and 3.2-3b would be revised to display only parameters RCS flow and power level in tables. The tables show required reduction in power level to account for reduced RCS total flow. The parameter R would be deleted since it is a constant value. This is an administrative change since the only change is in the presentation of these figures.

03-09

M

The allowed ACTIONS for reduced RCS flow would be deleted and replaced with the more restrictive ITS requirement to restore RCS flow within 2 hours.

03-10

LG

The note on measurement uncertainty for flow is moved to the Bases.

04-01

A

Clarifies that when the excore detectors are calibrated, the quadrant power tilt is zeroed out. (The QUADRANT POWER TILT RATIO (QPTR) is normalized to unity.) This requirement from NUREG-1431 as modified by Traveler TSTF-25, is consistent with the CTS ACTION requirements for verifying QPTR is within limit during power escalation subsequent to identifying and correcting the cause of QPTR out of limit.

04-02

LS10

The required CTS ACTION to calculate QPTR once per hour until THERMAL

CHANGE
NUMBER

NSHC

DESCRIPTION

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
		POWER was within limit or reduced to less than 50 percent RTP would be eliminated and replaced by new requirements from NUREG-1431. This represents a reduction in requirements for monitoring and reducing power.
04-03	LG	The details regarding obtaining QPTR using the incore detectors would be moved to the Bases.
04-04	LS12	<p>The CTS does not contain any provisions for determining QPTR with more than one inoperable input; thus, LCO 3.0.3 would be entered and the plant would be shut down. The proposed change would allow for the use of the movable incore detector system to determine an equivalent QPTR with one or more inoperable excore detector inputs to the QPTR calculation. In addition, the frequency is clarified by a note which says that the SR is not required until 12 hours after input from one or more power range channels is lost.</p> <p>If the movable incore detector system is used to determine an equivalent QPTR, the QPTR calculation is not based on information gained from any OPERABLE excore indications and therefore is independent of the number of OPERABLE excore detectors. The frequency specified in the CTS for the determination of an equivalent QPTR with the movable incore detectors (every 12 hours) would be retained. Further justification for this frequency is based on the fact that under normal circumstances, QPTR would not be expected to change significantly within a 12 hour period. If a significant change in QPTR were to occur, it would likely be the result of control rod misalignment which would be detectable immediately by means of the rod deviation monitor or rod bottom lights.</p>
04-05	LS11	ACTIONS requiring QPTR to be restored within 24 hours, QPTR to be verified during return to power, and to reset power range neutron flux-high trip setpoint to \leq 55 percent RTP would be eliminated. The ITS would add requirements for measuring $F_Q(Z)$ and $F_{\Delta H}^N$ (instead of QPTR) and performing safety analyses to verify peaking factors are acceptable prior to return to power.
04-06	LS13	ACTIONS involving QPTRs of 1.09 would be eliminated in conformance with NUREG-1431. While the requirements in CTS regarding QPTRs in excess of 1.09 due to misalignment of control rods would be addressed by the ITS requirements associated with rod group misalignment limits, the CTS ACTIONS regarding QPTRs in excess of 1.09 due to other causes would be replaced by less restrictive requirements. The CTS require that the QPTR be calculated once per hour and that power be reduced to less than 50 percent RTP within 2 hours and the power range neutron flux high Trip Setpoint be reduced within the next 4 hours. In addition, the CTS require identification and correction of the cause of the tilt condition and periodic verification that QPTR is within limits during any subsequent ascension to RTP. The ITS would require: (1) that QPTR be calculated only once per 12 hours, (2) only a 3 percent RTP reduction for each 1 percent of QPTR in excess of 1.0 and no reduction in flux trip setpoints, and (3) verification of peaking factors prior to and following power ascension and a reevaluation of safety analyses prior to power ascension. However, the requirements of the ITS are acceptable because: (1) the QPTR would be expected to change slowly over time so a less frequent calculation of QPTR would be acceptable; (2) once the operating staff commences a power reduction, in accordance with ITS requirements, the effect of any flux tilt will tend to be mitigated by reducing the flux and establishing greater margin to fuel design limits, and the reduction of power required by the ITS would result in a plant transient that generally would be less severe than the reduction to less than 50 percent as required by CTS. Further, eliminating the trip setpoint reduction is acceptable because a QPTR in excess of limits does not necessarily imply that accident analysis assumptions have been

CHANGE
NUMBER

NSHC

DESCRIPTION

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
		violated; and (3) the ITS required ACTIONS prior to and subsequent to power ascension provide assurance that POWER OPERATION at or near RTP will be in accordance with the safety analyses, and therefore acceptable.
04-07	A	The statement that Specification 3.0.4 does not apply is no longer needed as revised ACTIONS permit continued operation for an unlimited period of time.
04-08		Not used.
04-09	A	Consistent with NUREG-1431, a note is added to permit 3 OPERABLE excore channels to be used to calculate QPTR when 1 channel is inoperable and power is \leq 75 percent.
04-10	LS14	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-01	LG	The designation of how instrument uncertainties are treated (nominal, in the analysis or in the development of the TS limit) is moved to the Bases. The movement of this level of detail out of the specification is consistent with NUREG-1431 and is an example of removing unnecessary details from the TS in accordance with 10 CFR 50.36.
05-02	LS7	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-03	LG	Consistent with NUREG-1431, the requirement to perform a CHANNEL CALIBRATION on the RCS flow meters at least once per 18 months and the requirement to normalize the channels are moved to the Bases for the RCS flow low reactor trip function in ITS Section 3.3.1.
05-04	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-05	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-06	LS8	In accordance with NUREG-1431, if any of the DNB related parameters of pressure, temperature, or RCS flow are found to be outside their limits, the time period required to perform a power reduction would be extended to 6 hours. The DNB related parameters of RCS average temperature, pressurizer pressure, and RCS flow rate are maintained within specified limits in order to ensure consistency with the assumed initial conditions of the accident analyses. The limits placed on the RCS temperature, pressure, and flow ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed. Compliance with the above limits is verified every 12 hours. If a parameter is found to be outside the required limit, 2 hours are allowed in order to restore the parameter to within the limit. If the parameter is not restored to compliance within the required time, the plant must be shut down. The revised Completion Time of 6 hours is acceptable to allow transition to the required plant Conditions in an orderly manner without unnecessarily initiating any undue plant transients and on the small likelihood of a severe event occurring during the extended time period.
05-07	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-08		Not used.
05-09	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

**CHANGE
NUMBER**

NSHC

DESCRIPTION

05-10

A

The CTS requirement to verify RCS flow rate within limits prior to operation above 75 percent RTP after each fuel loading and at least every 31 effective full power days (EFPDs) would be eliminated from the ITS SRs for DNB parameters. This is acceptable based on the requirement to verify $F_{\Delta H}^N$ within limits prior to operation above 75 percent RTP after each fuel loading and every 31 EFPDs. Since the LCO, ACTION and SRs for $F_{\Delta H}^N$ would address any RCS flow rate problems, the RCS flow rate SR can be considered to duplicate the requirements of the $F_{\Delta H}^N$ ITS SR.

05-11

A

Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(8 pages)

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	Removes reference to the special test exception in accordance with the format of NUREG-1431.	Yes	Yes	Yes	Yes
01-02 LS1	The CTS allow 15 minutes to either restore AFD or reduce THERMAL POWER to less than 90% RTP. The ITS allow 15 minutes to restore AFD and an additional 15 minutes for the reduction in the THERMAL POWER if AFD cannot be restored within the original 15 minutes.	No, this is a CAOC requirement not in CTS.	Yes	No, this is a CAOC requirement not in CTS.	No, this is a CAOC requirement not in CTS.
01-03 LG	In accordance with Wolf Creek ITS for AFD, the details regarding how AFD is measured would be moved to the Bases.	No	No	Yes	No
01-04 M	The CTS allows the 16 hours of operation outside of the target band for surveillance testing of the power range neutron flux channels. In the ITS, the practical application is identified, and the defined surveillance testing only includes the incore/excore calibration.	No, this is a CAOC requirement not in CTS.	Yes	No, this is a CAOC requirement not in CTS.	No, this is a CAOC requirement not in CTS.
01-05 M	Additional requirements are imposed in the event reactor power is required to be reduced to less than or equal to 50% RTP due to accumulated AFD penalty minutes.	No, these requirements are not in CTS.	Yes	No, these requirements are not in CTS.	No, these requirements are not in CTS.
01-06 LS2	Eliminates the requirement to reduce the power range neutron flux - high reactor Trip Setpoints.	No, refer to 01-16-LS9.	Yes	No, refer to 01-16-LS9.	No, refer to 01-16-LS9.
01-07 LG	Moves additional surveillance frequencies, if an alarm is not OPERABLE, to licensee controlled documents. This change is consistent with TSTF 110.	Yes, to ECG and FSAR.	Yes, to TRM	Yes, to USAR Chapter 16.	Yes, to FSAR Chapter 16.
01-08	Not Used.	N/A	N/A	N/A	N/A
01-09 A	The description of when the AFD is considered to be outside its [limits] will be a note to the ITS LCO, instead of the CTS SR.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-10 A	Explicitly defines the surveillance frequency for determining the target AFD. In addition, a note is added which allows the use of a design target AFD after each refueling and prior to the initial determination of the target AFD.	No, target band requirement is not applicable to CTS.	Yes	No, target band requirement is not applicable to CTS.	No, target band requirement is not applicable to CTS.
01-11 A	For CPSES, deletes the explicit exception to CTS 4.0.4 for the AFD specification.	No	Yes	No	No
01-12 A	Applicability of the CTS is changed from greater than 50% RTP to greater than or equal to 50% RTP. The requirement to meet AFD limits prior to increasing power above 50% would be deleted. The applicability change is consistent with the AFD allowed operational space in the COLR, which is defined for 50% RTP and above. The elimination of the requirement to meet AFD requirements would have no impact on plant operations because it is understood that it is necessary to comply with TS requirements.	Yes	No, CPSES uses CAOC methodology. Change applies to RAOC methodology.	Yes	Yes
01-13 A	The ACTION statement regarding restoring AFD to within limits within 15 minutes would be deleted. This has no effect on the time allowed for completion of Required Actions and restoring AFD to within limits is implicit in requirements for exiting the ACTION statement. Therefore, this change is administrative.	Yes	No, CPSES uses CAOC methodology. Change applies to RAOC methodology.	Yes	Yes
01-14	Not Used.	N/A	N/A	N/A	N/A
01-15 M	The LCO, ACTION, and SRs associated with RAFDO would be eliminated. RAFDO is a methodology specific to Callaway.	No	No	No	Yes

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-16 LS9	Eliminates the requirement to reduce the power range neutron flux - high reactor Trip Setpoints when AFD is outside the RAOC limits.	Yes	No, CPSES uses CAOC methodology; change applies to RAOC methodology.	Yes	Yes
02-01 LG	The details of the entire CTS LCO of the allowable $F_o(Z)$ values, including the $K(Z)$ and $W(Z)$ parameters, would be moved to the COLR and/or Bases.	Yes	Yes	Yes	Yes
02-02 LS3	Consistent with Traveler TSTF-95, the Required Actions are relaxed to extend the time allowed to reduce the high neutron flux reactor Trip Setpoints [from 4 hours] to 72 hours.	Yes	Yes	Yes	Yes
02-03 M	The required ACTIONS are revised to include the addition of a requirement to be in at least MODE 2 within 6 hours should any of the ACTIONS not be completed within the required time.	Yes	Yes	Yes	Yes
02-04 M	$F_o^W(Z)$ must be verified to be within limits whenever $F_o(Z)$ is measured, not just at the time of target flux determination, as required by the CTS.	No, requirement not in CTS.	Yes	No, requirement not in CTS.	No, requirement not in CTS.
02-05 M	$F_o^C(Z)$ and $F_o^W(Z)$ must be verified to be within limits prior to exceeding 75% RTP after each refueling.	Yes	Yes	Yes	Yes
02-06 A	Consistent with the Bases of CTS 4.0.3, which allows 24 hours for completing SRs that become applicable when an exception to Specification 4.0.4 is allowed, the frequency for assessing $F_o(Z)$ is clarified by requiring that the measurement be performed within 24 hours after reaching equilibrium Conditions.	Yes	Yes	No, CLB will be retained.	No, CLB will be retained.

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-07 A	The footnote allowing power to be increased until the THERMAL POWER for extended operation has been achieved has been incorporated in the notes preceding ITS SR 3.2.1.1 and SR 3.2.2.1 for achieving equilibrium power level. This footnote replaces the Specification 4.0.4 exemption in the CTS.	Yes	Yes	Yes	Yes
02-08 LS4	The time allowed to reduce the acceptable operation limits on AFD is changed to 4 hours, consistent with Traveler TSTF- 99.	No, refer to 02-14-M.	Yes	Yes	Yes
02-09 M	The optional ACTION to comply with CTS 3.2.2 if $F_o^W(Z)$ exceeds its limit would be deleted. This eliminates an option and is more restrictive.	Yes	No, identified option not in CTS.	Yes	Yes
02-10	Not Used.	N/A	N/A	N/A	N/A
02-11 LS15	The ACTION requirement, with $F_o(Z)$ exceeding its limit, that the overpower Delta-T Trip Setpoint reduction be performed in at least HOT STANDBY would be deleted.	Yes	No, requirement not in CTS.	No, requirement not in CTS.	No, requirement not in CTS.
02-12 A	$F_o^W(Z)$ must be verified to be within limits whenever $F_o(Z)$ is measured as required by the CTS.	Yes	No, not required by CTS.	Yes	Yes
02-13 LG	The definition of extended operation (expected operation at a power level for greater than 72 hours) is moved to the Bases.	No, term not in CTS.	No, term not in CTS.	No, definition not in CTS.	Yes
02-14 M	Similar to CN 02-08-LS4. For DCP, there is currently no time limit; therefore, the imposition of a time limit is a more restrictive change.	Yes	No	No	No
03-01 LG	Moves the details of the $F_{\Delta H}^N$ limits to the COLR and/or Bases.	Yes	Yes	Yes	Yes
03-02 LS5	Revises the completion time to 4 hours (versus 2 hours in the CTS to attempt to restore $F_{\Delta H}^N$ to within its limit or to reduce power to below 50% RTP.	Yes	Yes	No, 4 hours already in CTS.	Yes

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-03 M	The requirement to reduce power to less than or equal to 5% RTP (exit MODE 1) within the next 6 hours is added in lieu of the use of CTS 3.0.3.	Yes	Yes	Yes	Yes
03-04 LS6	Revises the time allowed to reduce the reactor power to a Condition where the LCO does not apply (MODE 2, < 5% RTP) to 6 hours (versus 2 hours in the CTS).	Yes	Yes	No, 6 hours already in CTS.	Yes
03-05 M	Requires $F_{\Delta H}$ ACTION Statements to be completed if entered.	Yes	Yes	Yes	Yes
03-06 A	A note would be added to state that THERMAL POWER does not need to be reduced in order to comply with the series of flux maps that must be taken upon a return to power.	Yes	Yes	Yes	Yes
03-07 A	The DCPD specific parameter R, which is a derived value based on $F_{\Delta H}^N$, would be deleted.	Yes	No	No	No
03-08 A	Revise DCPD specific Figures 3.2-3a and 3.2-3b to display only parameters RCS flow and power level. The Parameter R would be deleted since it is a constant value.	Yes	No	No	No
03-09 M	The DCPD specific allowed ACTIONS for reduced RCS flow would be deleted and replaced with the more restrictive ITS requirement to restore RCS flow within 2 hours.	Yes	No	No	No
03-10 LG	The DCPD note on measurement uncertainty for flow is moved to the Bases.	Yes	No	No	No
04-01 A	Clarifies that when the excore detectors are calibrated, the quadrant power tilt is zeroed out. (The QPTR is normalized to unity.)	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
04-02 LS10	The required CTS ACTION to calculate QPTR once per hour until THERMAL POWER was within limit or reduced to less than 50% RTP would be eliminated and replaced by new requirements from NUREG-1431. This represents a reduction in requirements for monitoring and reducing power.	Yes	No, requirement not in CTS.	Yes	Yes
04-03 LG	The details regarding obtaining QPTR using the incore detectors would be moved to the Bases.	Yes	No, details not in CTS.	Yes	Yes
04-04 LS12	The requirements and capabilities for measuring QPTR when 1 or more excore detector channels are inoperable are clarified.	Yes	Yes, see also change 01-29-LS in Section 1.0.	No, see CN 04-10-LS14.	Yes
04-05 LS11	CTS ACTIONS requiring QPTR to be restored within 24 hours, QPTR to be verified during return to power, and power range neutron flux-high trip setpoint to be reset to $\leq 55\%$ would be eliminated.	Yes	No, requirement not in CTS.	No, see CN 04-10-LS14.	Yes
04-06 LS13	CTS ACTIONS involving QPTR exceeding 1.09 would be eliminated in conformance with NUREG-1431.	Yes	No, actions not in CTS.	Yes	Yes
04-07 A	The statement that Specification 3.0.4 does not apply is no longer needed as revised ACTIONS permit continued operation for unlimited period of time.	Yes	No, exception not in CTS.	Yes	Yes
04-08 A	Not Used.	N/A	N/A	NA	N/A
04-09 A	Consistent with NUREG-1431, a note is added to permit 3 OPERABLE excore channels to be used to calculate QPTR when 1 channel is inoperable and power is $\leq 75\%$.	Yes	No, already in CTS.	No, maintaining CTS wording.	Yes
04-10 LS14	The allowed time for the requirement to reset the power range neutron flux-high setpoint during power reduction required by QPTR ACTIONS would be extended to 72 hours for Wolf Creek.	No	No	Yes	No

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-01 LG	The designation of how instrument uncertainties are treated (nominal, in the analysis, or in the development of the CTS limit) is moved to the Bases.	Yes	Yes	Yes	Yes
05-02 LS7	The CPSES specific requirement to verify that the total RCS flow is within limits using the plant computer or elbow tap output voltage on a monthly basis is deleted.	No	Yes	No	No
05-03 LG	The requirement to perform a CHANNEL CALIBRATION at least once per 18 months and the requirement to normalize the channels are moved to the Bases for the SRs for the RCS flow - low reactor trip function in ITS 3.3.1.	Yes	Yes	Yes	Yes
05-04 LG	Consistent with industry Traveler TSTF-105, the explicit requirements that the RCS flow be measured through the use of a precision heat balance measurement and that the instrumentation used in the performance of the calorimetric flow measurement be calibrated within a specified time period of performing the measurement is moved to the Bases.	No, requirement not in CTS.	Yes	Yes	Yes
05-05 LG	The Wolf Creek Required ACTIONS would be modified to move details regarding identification of the cause for low flow rate to the Bases.	No	No	Yes	No
05-06 LS8	The time to reduce power to less than 5% RTP would be revised from within 4 hours to within the next 6 hours.	Yes	Yes	Yes	Yes
05-07 M	This surveillance is modified for Callaway to require that it be performed within 7 days of achieving 95% RTP.	No, see ITS Section 3.4, CN 3.4-51.	No, see CN 5-11-A.	Yes	Yes
05-08	Not Used.	N/A	N/A	N/A	N/A
05-09 LG	The requirements for inspecting and cleaning the feedwater flow venturi would be moved to licensee-controlled documents.	No, requirement not in CTS.	No, requirement not in CTS.	Yes, to USAR, Chapter 16.	Yes, to FSAR, Chapter 16.

CONVERSION COMPARISON TABLE - CTS 3/4.2

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-10 A	The CTS requirement to verify RCS flow rate within limits prior to operation above 75% RTP after each fuel loading and at least every 31 EFPDs would be eliminated from the SRs for departure from nucleate boiling (DNB) parameters.	Yes	No, requirement not in CTS.	No, requirement not in CTS.	Yes
05-11 A	<p>The change is specific to Comanche Peak. An ACTION is added to clarify that the accident analyses support operation below 85% RTP with a reduced flow rate, and this Condition is not affected by the failure of a precision RCS flow measurement to verify that the required flow exists.</p> <p>In addition, the parameters to be verified per the LCO are clarified. Failure of the precision flow measurement when below 85% RTP following a refueling outage does not result in the violation of the LOC; it only prohibits POWER ASCENSION above 85% RTP.</p>	No	Yes	No	No

ENCLOSURE 4
NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, AI, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3...." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R" (Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility, involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3

10CFR50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

With the heat flux hot channel factor (F_o) or the enthalpy rise hot channel factor ($F_{\Delta H}^N$) outside their respective limits, the CTS require the THERMAL POWER and the power range neutron flux - high reactor Trip Setpoints be reduced. The CTS allow [4] hours for the completion of the setpoint reduction. As written, the Completion Time presents an unnecessary burden on the operation of the plant. A Completion Time of 72 hours will allow time to perform a second flux map to confirm the results, or determine that the Condition was temporary, without implementing an unnecessary trip setpoint change. During the Trip Setpoint change, there is increased potential for human error resulting in a plant transient. Following a significant power reduction, at least 24 hours required to reestablish steady-state xenon Conditions prior to taking a flux map, and approximately 12 hours to obtain the flux map and analyze the data. An increased potential for inadvertent reactor trip can be created through requiring the Trip Setpoints to be reduced within the same time frame that a unit power reduction is taking place, and within the current Completion Time. Setpoint adjustment is estimated to take approximately 4 hours per channel (review of plant Conditions supportive of removing channels from service, tripping of bistables, setpoint adjustments, and channel restoration), adding 2 hours for necessary initial preparations (procedure preparations, calibration equipment checks, obtaining tools and approvals), it is reasonable to expect a total of 18 hours. Further, setpoint changes should only be required for extended operation in this Condition. Finally, the basis for making this setpoint change are the same as the NUREG-1431 Bases provided for the 72 hour Completion Time of LCO 3.2.1, Required ACTION A.4., which is also a setpoint reduction. Therefore, a Completion Time of 72 hours for any required setpoint change maintains consistency within the ITS.

The proposed change, consistent with industry Traveler TSTF95, would allow 72 hours for the completion of the setpoint reduction. The reduction of the setpoints is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the amount of work required to be done to reduce the setpoints, the small likelihood of a severe transient in this time period, and the prompt reduction in THERMAL POWER required upon discovery of the out-of-limit Condition.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not involve any new operating activities or hardware changes; thus the proposed change has no effect on the probability of an accident.

Because the accident analyses are initiated from within the Conditions defined by the TS LCO and these LCOs are unchanged, the accident analyses are unaffected. This change extends the time allowed for a

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3 (continued)

conservative ACTION to be taken. The time extension allows the reactor Trip Setpoint reduction to be performed in a more deliberate manner, thereby, reducing the potential for inadvertent reactor trips introduced during the setpoint reduction. Furthermore, the prompt reduction in THERMAL POWER required upon discovery of the out-of-limit Condition restores the margins of the accident analyses for those transients which do not involve positive reactivity excursions. Due to the small likelihood of an event during the additional delay in the time allowed to reset the reactor Trip Setpoints, there is not a significant effect on the consequences of an accident previously analyzed.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the proposed increase in the time allowed to reduce the reactor Trip Setpoints. Therefore, there is no possibility for a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The accident analyses are assumed to be initiated from conditions which are consistent with the TS LCO. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no effect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, ITS is concluded that the activities associated with NSHC "LS3" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c) and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5

10 CFR 50.92 EVALUATION FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

With the nuclear enthalpy rise hot channel factor ($F_{\Delta H}^N$) outside its limits, the CTS allow 2 hours to restore the $F_{\Delta H}^N$ to within its limits prior to initiating other actions. The proposed change, consistent with NUREG-1431, would allow 4 hours for the same activities to be completed. The peaking factor may be brought to within its limits by rod realignment or by reducing the power to within the power-dependent $F_{\Delta H}^N$. The completion time of 4 hours provides an acceptable time to restore $F_{\Delta H}^N$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time and without imposing any rapid power reduction transients on the plant.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not involve any new operating activities or hardware changes; thus the proposed change has no effect on the probability of an accident.

Because the accident analyses are initiated from within the conditions defined by the TS LCO and these LCOs are unchanged, the accident analyses are unaffected. This change extends the time allowed for an action to be taken. The time extension allows for power reductions to be performed in a more deliberate manner; thereby, reducing the potential for inadvertent transients initiated during the power reduction. Furthermore, the prompt action to reduce reactor power places the plant in a better situation to withstand a transient without any increased consequences. Therefore, it is concluded that the additional time allowed to reduce the reactor Trip Setpoints, after the initial power reduction, does not result in a significant increase in the consequences of any accident previously analyzed.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the proposed increase in the time allowed to bring the $F_{\Delta H}^N$ to within its limits. Therefore, there is no possibility for a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The accident analyses are assumed to be initiated from conditions which are consistent with the TS LCO. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no effect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5
(continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS5" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

With the nuclear enthalpy rise hot channel factor ($F_{\Delta H}^N$) having been outside its limits, the TS require that within 24 hours after exceeding the $F_{\Delta H}^N$ limit, an incore flux map be performed to verify that the $F_{\Delta H}^N$ has been restored to within its limits. If this activity is not completed, the CTS require that the plant be taken to MODE 2 within the next 2 hours. The proposed change, consistent with NUREG-1431, would allow 6 hours for the same activity to be completed. This time frame is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power Conditions in an orderly manner and without challenging plant systems. The LCO is not applicable in MODE 2.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not involve any new operating activities or hardware changes; thus the proposed change has no effect on the probability of an accident.

Because the accident analyses are initiated from within the conditions defined by the TS LCO and these LCOs are unchanged, the accident analyses are unaffected. This change extends the time allowed for an action to be taken. The time extension allows the power reduction to MODE 2 to be performed in a more deliberate manner; thereby, reducing the potential for inadvertent transients initiated during the power reduction and minimizing challenges to the plant systems. However, due to the small likelihood of a transient during the additional time period and because the reactor operators are taking positive steps to either comply with the specification or to reduce reactor power, the consequences of the accident analyses are not significantly affected by this change.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the proposed increase in the time allowed to bring to MODE 2. Therefore, there is no possibility for a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The accident analyses are assumed to be initiated from conditions which are consistent with the TS LCO. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no effect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6
(continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS6" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The DNB related parameters of RCS average temperature, pressurizer pressure, and RCS flow rate are maintained within specified limits in order to ensure consistency with the assumed initial Conditions of the accident analyses. The limits placed on the RCS temperature, pressure, and flow ensure that the minimum departure from DNBR will be met for each of the transients analyzed. Compliance with the above limits is verified every 12 hours. If a parameter is found to be outside the required limit, 2 hours are allowed in order to restore the parameter to within the limit. If the parameter is not restored to compliance within the required time, the plant must be shut down.

This proposed change would extend the time allowed for the required plant shutdown from 4 hours to 6 hours. The Completion Time of 6 hours is reasonable to reach the required plant Conditions in an orderly manner and without unnecessarily initiating any undue plant transients.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not involve any new operating activities or hardware changes; thus the proposed change has no effect on the probability of an accident.

Because the accident analyses are initiated from within the Conditions defined by the TS LCO and these LCOs are unchanged, the accident analyses are unaffected. The occurrence of a DNB challenging event while degraded DNB parameters existed would be expected to affect the consequences of the event. However, the proposed change would not increase the probability of an accident or transient and the additional time to shut the plant down would tend to avoid inadvertent transients. This change extends the time in which the plant is required to be shutdown from 4 hours to 6 hours. This relatively small increase in the shutdown time allows for a more orderly plant shutdown which minimizes the potential for initiating another, unnecessary plant transient during the shutdown process. The 2-hour extension to perform the shutdown does not result in any significant increase in the consequences due to the small likelihood of an event during this time frame and the orderly power reduction in progress. Hence, there will be no effect on any of the accident analysis assumptions and the consequences of the accident analyses are unaffected by this change.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the proposed extension of the time required to shut down the plant. Therefore, there is no possibility for a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change only affects the Completion Time required to shut down the plant if selected DNB parameters are not within limits. The proposed change does not affect any LCO. Because the accident analyses are assumed to be initiated from conditions which are consistent with the TS LCO, the proposed change does not affect any of the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no affect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS8" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

With the AFD not within the limits specified in the COLR, the CTS required that the THERMAL POWER be reduced to less than 50 percent RTP and that the power range neutron flux - high reactor Trip Setpoints be reduced to less than or equal to 55 percent RTP within the next 4 hours. The requirement to reduce the high neutron flux reactor Trip Setpoints is proposed to be deleted in conformance with NUREG-1431. The proposed change is considered to be acceptable because reducing the power level to < 50 percent RTP maintains the plant in a Condition where the AFD is not a significant accident analysis input and the probability of an accident or transient that would cause an increase in reactor power, for which reduced Trip Setpoints would provide additional protection, is very low.

There are no AFD limits below 50 percent RTP. A rise to \geq 50 percent RTP with AFD outside the limits does not immediately create an unacceptable situation, and a reactor trip or timely operator action depending on the rapidity of the power transient, would successfully terminate the event. Rapid power excursions resulting from events that are peaking factor limiting would likely be terminated by normal reactor trip signals or reactor trip on safety injection. Slower transients would be terminated by the operator or, for large AFD deviations, the overtemperature ΔT reactor trip. The overtemperature ΔT reactor Trip Setpoint is automatically reduced when AFDs deviate sufficiently from the required operating area. Thus, reducing the high neutron flux setpoint provides an additional level of accident mitigation which would not be necessary in most cases.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves only the compensatory measures required for the Condition where AFD is outside limits; it does not involve any new operating activities or hardware changes. Thus the proposed change has no effect on the probability of an accident.

Since the reactor trip system and operator action will continue to provide an adequate level of protection for transients involving Conditions where AFD is not within limits, the proposed change would have an insignificant effect on the results of accidents or transients. Therefore, the proposed change will have an insignificant effect on the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the proposed deletion of an overly restrictive reactor Trip Setpoint reduction. Therefore, there is no possibility for a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change retains the requirement to reduce power to < 50 percent RTP within 30 minutes. This provides an acceptable level of protection when AFD is not within limits. The REACTOR TRIP SYSTEM also would continue to provide an adequate level of protection against an unlikely accident and transient that may occur. Further, resetting the Trip Setpoints involves some risk of causing a plant trip and consequential transient. Therefore, the proposed change would not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS9" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The required CTS ACTION to calculate QPTR once per hour until THERMAL POWER was reduced to less than 50 percent RTP would be eliminated and replaced by new requirements from NUREG-1431. This represents a reduction in requirements for monitoring and reducing power. With QPTR not within limit, the CTS require a calculation of QPTR at least once per hour until either the QPTR is restored or THERMAL POWER is reduced to < 50 percent RTP. The comparable ITS ACTIONS would require QPTR to be calculated at least once per 12 hours and continue to reduce THERMAL POWER by at least 3 percent for each 1 percent that QPTR exceeds 1.00 until either QPTR is restored to within its limit or 50 percent RTP is achieved. The 12-hour frequency is sufficient because, as stated in the NUREG-1431 Bases, further changes in QPTR would be relatively slow. The once per hour frequency is excessive considering the slow rate of flux change and potentially would divert the attention of control room staff from corrective action with respect to QPTR.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves only the compensatory measures to be taken should the QPTR be outside its limit. The frequency with which QPTR is calculated is not assumed in the initiating events for any accident previously evaluated. In addition, the change does not involve any new operating activities or hardware changes. Therefore, the proposed change would not significantly increase the probability of an accident previously evaluated.

Once THERMAL POWER has been reduced appropriately in proportion to the amount that QPTR exceeds 1.0, any additional change would be sufficiently slow that a 12-hour interval for recalculating QPTR will provide an adequate level of protection. Therefore, the proposed change will not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not involve a physical alteration to the plant; no new or different kind of equipment will be installed. Also, the manner in which the plant would be operated would not be altered. Thus, the change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will continue to ensure that the plant is maintained in safe condition while QPTR is in excess of its limit. Additionally, calculating QPTR once per 12 hours as opposed to every hour while

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10 (continued)

QPTR is in excess of its limit would avoid the diversion of personnel resources from corrective actions with regard to meeting the LCO. Therefore, the proposed change will not involve a significant reduction in any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS10" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS11

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

CTS ACTIONS requiring QPTR to be restored within 24 hours or reduce power to < 50 percent RTP and requiring verification of QPTR during return to full power operation would be eliminated in accordance with NUREG-1431. Also, the requirement to reset power range neutron high-flux Trip Setpoint during the power reduction and after a required reduction to \leq 50 percent RTP would be eliminated. The ITS would add requirements for measuring $F_0(Z)$ and $F_{\Delta H}^N$ (instead of QPTR) and performing safety analyses to verify peaking factors are acceptable prior to return to power.

The ITS focus on maintaining the peaking factors $F_0(Z)$ and $F_{\Delta H}^N$ within limits rather than the QPTR. This is appropriate because QPTR is a monitored parameter that is indicative of peaking factor problems. The ITS require verification that $F_0(Z)$ and $F_{\Delta H}^N$ are within limits within 24 hours by performing SRs that can directly measure flux shapes in the core. If $F_0(Z)$ or $F_{\Delta H}^N$ are not within limits, the Conditions for those TS will specify the Required ACTIONS. Since the peaking factors are of prime importance, the ITS will ensure that the power distribution remains consistent with the initial conditions assumed in safety analyses. The proposed Completion Time takes into consideration the rate at which peaking factors are likely to change and the time required to stabilize the plant and perform a flux map.

The ITS would retain the 2-hour requirement to reduce power proportionally to the percent that QPTR exceeds its limit. This would result in a power reduction that would provide additional margin to fuel design limits during a flux tilt Condition to assure that design limits are not challenged by local flux peaking. These design margins are set conservatively and provide further assurance that operation during the 24-hour period would not challenge fuel design limits.

The ITS would require a reevaluation of the safety analyses prior to increasing reactor power above the reduced power required by the QPTR limit. Finally, the ITS would require a confirmation that $F_0(Z)$ and $F_{\Delta H}^N$ are within limit following the power increase.

The proposed changes also would eliminate the requirements to reset the power range neutron flux -- high Trip Setpoints. First, the requirement to reduce the setpoints within 4 hours following power reductions proportional to the percent QPTR exceeds the limit would be eliminated. Second, the requirement to reduce the setpoints to \leq 55 percent RTP within 4 hours of reaching 50 percent RTP would be eliminated. The former change is acceptable on the basis that the likelihood of an event occurring during the power reduction phase and during the 24 hour period prior to verifying peaking factors within limits is small. The latter change is acceptable on the basis that the ITS would require peaking factors to be determined in the same time frame as the CTS, and the peaking factor ITS have their own requirements, with appropriate Completion Times, for reducing reactor power and resetting the power range neutron flux -- high Trip Setpoints.

This proposed TS changes have been evaluated and it has been determined that they involve NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21.(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS11 (continued)

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves only the compensatory measures to be taken should the QPTR be outside its limit. These compensatory measures are not assumed in the initiating events for any accident previously evaluated. Therefore, the proposed change will not affect the probability of any accident previously evaluated. Furthermore, the proposed changes to these compensatory measures, which are derived from NUREG-1431, would continue to provide acceptable levels of protection. Therefore, the proposed changes will not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not involve a physical alteration to the plant; no new or different kinds of equipment would be installed. The changes would not alter the manner in which the plant would be operated. Thus, the changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed changes will continue to ensure that the plant is maintained in a safe condition within the envelope of the safety analyses while QPTR is in excess of its limit. Therefore, the proposed changes will not involve a significant reduction in any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS11" resulting from the conversion to the ITS format satisfy the NSHC of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS12

10 CFR 50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The QPTR is defined as the ratio of the maximum of the four excore detector calibrated output to the average of the four excore detector calibrated outputs for the upper half of the detectors and the lower half of the detectors. If 1 of the excore detector inputs to the QPTR calculation becomes inoperable, the CTS allow the use of the moveable incore detector system to determine an equivalent QPTR. [If, at or below 75 percent RTP, one of the excore detector inputs to the QPTR calculation becomes inoperable, the CTS allow the use of the remaining three detectors to determine an equivalent QPTR.] Further, if the moveable incore detector system is used to determine an equivalent QPTR, the current Specifications do not contain any provisions for determining QPTR with more than one inoperable unit; thus LCO 3.0.3 would be entered and the plant would be shut down.

The proposed change would allow for the use of the movable incore detector system to determine an equivalent QPTR with 1 or more inoperable excore detector inputs to the QPTR calculation. If the movable incore detector system is used to determine an equivalent QPTR, the QPTR calculation is not based on information gained from any operable excore indications, and therefore is independent of the number of operable excore detectors. The frequency specified in the CTS for the determination of an equivalent QPTR with the movable incore detectors (every 12 hours) would be retained. The frequency is clarified by a note which says that the SR is not required until 12 hours after input from 1 or more power range channels become inoperable. Further justification for this frequency is based on the fact that under normal circumstances, QPTR would not be expected to change significantly within a 12 hour period. If a significant change in QPTR were to occur, it would likely be the result of control rod misalignment which would be detectable immediately by means of the rod deviation monitor or rod bottom lights.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not involve any new operating activities or hardware changes; thus the proposed change has no effect on the probability of an accident.

This change makes available an option to ensure that continued plant operation (quadrant power tilt) is within the assumptions of the accident analyses without imposing an unnecessary transient on the plant. The limits on the quadrant power tilt ratio LCO are unchanged. Because the accident analyses are initiated from within the Conditions defined by the TS LCOs, and these LCOs are unchanged, the accident analyses are unaffected. Therefore, there will be no effect on any of the accident analysis assumptions and the consequences of the accident analyses are unaffected by this change.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS12 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the proposed alternate method of determining an equivalent QPTR with more than 1 inoperable excore detector inputs. Therefore, there is no possibility for a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The accident analyses are assumed to be initiated from conditions which are consistent with the TS LCO. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no affect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS12" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO HAZARDS CONSIDERATIONS

NHSC LS13

10 CFR 50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

ACTIONS involving QPTRs or 1.09 would be eliminated in conformance with NUREG-1431. While the requirements in CTS regarding QPTRs in excess of 1.09 due to misalignment of control rods would be addressed by the ITS requirements associated with rod group misalignment limits, the CTS ACTIONS regarding QPTRs in excess of 1.09 due to other causes would be replaced by less restrictive requirements. The CTS require that the QPTR be calculated once per hour and that power be reduced to less than 50 percent RTP within 2 hours and the power range neutron flux high Trip Setpoint be reduced within the next 4 hours. In addition, the CTS require identification and correction of the cause of the tilt condition and periodic verification that QPTR is within limits during any subsequent ascension to RTP. The ITS would require: (1) that QPTR be calculated only once per 12 hours, (2) only a 3 percent RTP reduction for each 1 percent of QPTR in excess of 1.0 and no reduction in flux trip setpoints, and (3) verification of peaking factors prior to and following power ascension and a reevaluation of safety analyses prior to power ascension.

The proposed changes are acceptable because:

- (1) The QPTR would be expected to change slowly over time so a less frequent calculation of QPTR would be acceptable;
- (2) Once the operating staff commences a power reduction in accordance with its ITS requirements, the effect of any flux tilt will tend to be mitigated by reducing the flux and establishing greater margin to fuel design limits.

The reduction of power required by the ITS would result in a plant transient that generally would be less severe than the reduction to less than 50 percent as required by CTS, and eliminating the trip setpoint reduction is acceptable because a QPTR in excess of limits does not necessarily imply that accident analysis assumptions have been violated; and

- (3) the ITS Required ACTIONS prior to and subsequent to power ascension provide assurance that power operation at or near RTP will be in accordance with the safety analyses, and therefore acceptable.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated, or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated, or Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes do not involve any new methods of operating the plant or hardware changes; thus the proposed change has no effect on the probability of an accident.

The proposed change involves only the compensatory measures to be taken should the QPTR be outside its limit. These compensatory measures are not assumed in the initiating events for any accident

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS13 (continued)

previously evaluated. Therefore, the proposed changes will not affect the probability of any accident previously evaluated. Furthermore, the proposed changes to these compensatory measures, which are derived from NUREG-1431, would continue to provide acceptable levels of protection. Therefore, there will be no effect on any of the accident analysis assumptions and the consequences of the accident analyses are unaffected by this change.

Therefore, the proposed changes will not significantly increase the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not involve a physical alteration to the plant; no new or different kinds of equipment would be installed. The changes would not alter the manner in which the plant would be operated. Thus, the changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change involves only the compensatory measures to be taken should the QPTR be outside its limit. The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the proposed alternate methods of dealing with QPTRs in excess of 1.09. Therefore, there is no possibility for a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The proposed changes, which involve implementing NUREG-1431 requirements, will continue to ensure that the plant is maintained in a safe condition within the envelope of the safety analyses while QPTR is in excess of its limit. While different actions are taken in response to QPTR in excess of 1.09, the proposed changes would assure that accident analysis assumptions continue to be met. Therefore, the proposed changes will not involve a significant reduction in any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS13" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NHSC LS15

10 CFR 50.92 EVALUATION FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

With the heat flux hot channel factor ($F_o(z)$) outside its limit, the CTS require that THERMAL POWER be reduced at least 1 percent for each 1% $F_o(z)$ exceeds the limit within 15 minutes and that the power range neutron flux - high trip setpoints be reduced within the next 4 hours. POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the overpower ΔT Trip Setpoints have been reduced at least 1 percent for each 1 percent $F_o(z)$ exceeds the limit. The CTS require that the overpower ΔT Trip Setpoint reduction be performed with the reactor in at least HOT STANDBY. The proposed change, consistent with NUREG-1431, would delete the requirement that the reactor be in at least HOT STANDBY while performing the overpower ΔT Trip Setpoint reduction. It is sufficient to reduce power 1 percent for each 1 percent $F_o(z)$ exceeds its limit and then perform the required Trip Setpoint reduction at reduced power. The CTS requirement to be in at least HOT STANDBY is not a Westinghouse design basis requirement.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not involve any new operating activities or hardware changes; thus the proposed change has no effect on the probability of an accident.

Because the accident analyses are initiated from within the conditions defined by the TS LCO and these LCOs are unchanged, the accident analyses are unaffected. This change removes from the TS the requirement that the plant be in at least HOT STANDBY while performing the overpower ΔT Trip Setpoint reduction. The magnitude of the setpoint reduction would remain unchanged and the maximum permissible THERMAL POWER before and after the setpoint reduction would remain unchanged. Hence, there will be no effect on any of the

IV. SPECIFIC NO HAZARDS CONSIDERATIONS

NHSC LS15 (continued)

accident analysis assumptions and the consequences of the accident analyses are unaffected by this change.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the proposed deletion of the requirement to be in at least HOT STANDBY while performing the setpoint reduction. Therefore, there is no possibility for a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The accident analyses are assumed to be initiated from Conditions which are consistent with the TS LCO. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no effect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS15" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers (1 Page)

NUREG-1431 Specifications that are not applicable (1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
3.2.1.....	3.2-1
3.2.2.....	3.2-6
3.2.3.....	3.2-9
3.2.4.....	3.2-10

Methodology (2 Pages)

Industry Travelers Applicable to Section 3.2

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-24	Not Incorporated	NA	Not NRC approved as of traveler cut-off date
TSTF-95	Incorporated	3.2-06	Approved by NRC.
TSTF-97	Incorporated	3.2-07	Approved by NRC.
TSTF-98, R1	Incorporated	3.2-03	
TSTF-99	Incorporated	3.2-08	Approved by NRC.
TSTF-109	Incorporated	3.2-15	Approved by NRC.
TSTF-110, R1	Incorporated	3.2-10	
TSTF-112, R1	Not Incorporated	NA	Not NRC approved as of traveler cut-off date
TSTF-136	Incorporated	NA	
TSTF-164	Incorporated	3.2-11	Applicable to CAOC only. (CPSES)
WOG-95, proposed Rev. 2	Incorporated	3.2-05/3.2-18	
WOG-105	Incorporated	3.2-16	

NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

<u>Specification #</u>	<u>Specification Title</u>	<u>Comments</u>
3.2.1A	Heat Flux Hot Channel Factor ($F_q(Z)$) (F_{xy} Methodology)	Not Applicable for $F_q(Z)$ Methodology Plants
3.2.3A	AXIAL FLUX DIFFERENCE (AFD) (Constant Axial Offset Control (CAOC) Methodology)	Not Applicable for Relaxed Axial Offset Control (RAOC) Methodology Plants

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_0(Z)$) (~~F_0 -Methodology~~)

LCO 3.2.1 $F_0(Z)$, as approximated by $F_0^c(Z)$ and $F_0^h(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. $F_0^c(Z)$ not within limit.</p>	<p>A.1 Reduce THERMAL POWER $\geq 1\%$ RTP for each 1% $F_0^c(Z)$ exceeds limit.</p>	<p>15 minutes</p>
	<p><u>AND</u></p>	
	<p>A.2 Reduce Power Range Neutron Flux-High trip setpoints $\geq 1\%$ for each 1% $F_0^c(Z)$ exceeds limit.</p>	<p>72 <u>8</u> hours <u>3.2-06</u></p>
	<p><u>AND</u></p>	<p>72 hours</p>
	<p>A.3 Reduce Overpower ΔT trip setpoints $\geq 1\%$ for each 1% $F_0^c(Z)$ exceeds limit.</p>	<p>Prior to increasing THERMAL POWER above the limit of Required Action A.1</p>
	<p><u>AND</u> A.4 Perform SR 3.2.1.1.</p>	

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. $F_0^W(Z)$ not within limits.	B.1 Reduce AFD limits $\geq 1\%$ for each 1% $F_0^W(Z)$ exceeds limit.	4 2 hours <u>3.2-08</u>
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----

During power escalation at the beginning of each cycle following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

3.2-13

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify F ₀ ^c (Z) is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within [12] 24 <u>3.2-12</u> hours after achieving equilibrium conditions after exceeding, by <u>3.2-02</u> ≥ 10% 20% RTP, the THERMAL POWER at which F ₀ ^c (Z) was last verified <u>AND</u> 31 EFPD thereafter

(continued)

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	<p>Once within <u>3.2-12</u> [12] 24 hours after achieving equilibrium conditions after exceeding, by $\geq 10\%$ 20% RTP, the THERMAL POWER at which $F_0^W(Z)$ <u>3.2-02</u> was last verified</p> <p><u>AND</u></p> <p>31 EFPD thereafter</p>

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor (F^N_{ΔH})

LCO 3.2.2 F^N_{ΔH} shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ----- F^N_{ΔH} not within limit.</p>	<p>A.1.1 Restore F^N_{ΔH} within limit.</p>	<p>4 hours</p>
	<p><u>OR</u></p>	<p>4 hours</p>
	<p>A.1.2.1 Reduce THERMAL POWER to < 50% RTP.</p>	<p>72 hours</p>
	<p><u>AND</u> A.1.2.2 Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP.</p>	<p><u>3.2-06</u></p>
	<p><u>AND</u> A.2 Perform SR 3.2.2.1. <u>AND</u></p>	<p>24 hours (continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 -----NOTE----- THERMAL POWER does not have to be reduced to comply with this Required Action: ----- Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP <u>AND</u> Prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----

During power escalation at the beginning of each cycle following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

3.2-13

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify F _{ΔH} ^N is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LC0 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

-----NOTE-----
 The AFD shall be considered outside limits when two or more
 OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER \geq 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify AFD within limits for each OPERABLE excore channel.	7 days AND <u>3.2-10</u> Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be \leq 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. QPTR not within limit.</p>	<p>A.1 Reduce THERMAL POWER \geq 3% from RTP for each 1% of QPTR > 1.00.</p>	<p>2 hours</p>
	<p><u>AND</u></p> <p>A.2 Determine QPTR Perform SR 3.2.4.1 and reduce THERMAL POWER \geq 3% from RTP for each 1% of QPTR > 1.00.</p>	<p>Once per 12 hours</p> <p style="text-align: right;"><u>3.2-15</u></p>
	<p><u>AND</u></p> <p>A.3 Perform SR 3.2.1.1 SR 3.2.1.2 and SR 3.2.2.1.</p>	<p>24 hours</p> <p style="text-align: right;"><u>3.2-16</u></p> <p><u>AND</u></p> <p>Once per 7 days thereafter</p>
	<p><u>AND</u></p> <p>A.4 Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.</p> <p><u>AND</u></p>	<p>Prior to increasing THERMAL POWER above the limit of Required Actions A.1 and A.2</p> <p style="text-align: right;"><u>ED</u></p> <p style="text-align: right;">(continued)</p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.5 -----NOTE----- Perform Required Action A.5 only after Required Action A.4 is completed. -----</p> <p>Calibrate Normalize excure detectors to eliminate tilt show zero QPTR.</p> <p>AND</p> <p>A.6 -----NOTE----- Perform Required Action A.6 must be completed when only after Required Action A.5 is implemented completed. -----</p> <p>Perform SR 3.2.1.1 SR 3.2.1.2 and SR 3.2.2.1.</p>	<p>Prior <u>3.2-05</u> to <u>ED</u> increasi ng THERMAL POWER above the limit of Required Actions A.1 and A.2</p> <p>Within <u>3.2-16</u> 24 hour s after reaching RTP</p> <p>OR</p> <p>Within <u>ED</u> 48 hours after increasing THERMAL POWER above the limit of Required Actions A.1 and A.2</p>
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to \leq 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.4.1 -----NOTES-----</p> <p>1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER < 75% RTP, the remaining three power range channels can be used for calculating QPTR.</p> <p>2. SR 3.2.4.2 may be performed in lieu of this Surveillance if adequate Power Range Neutron Flux channel inputs are not OPERABLE.</p> <p>-----</p> <p>Verify QPTR is within limit by calculation.</p>	<p><u>3.2-09</u></p> <p><u>3.2-15</u></p> <p>7 days</p> <p><u>AND</u> <u>3.2-10</u></p> <p>Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable</p>
<p>SR 3.2.4.2 -----NOTE-----</p> <p>Not Only required to be performed until 12 hours after if the input from one or more Power Range Neutron Flux channels are is inoperable with THERMAL POWER ≥ 75% RTP.</p> <p>-----</p> <p>Verify QPTR is within limit using the movable incore detectors.</p>	<p><u>3.2-15</u></p> <p><u>3.2-09</u></p> <p>Once within 12 hours</p> <p><u>AND</u> <u>3.2-15</u></p> <p>12 hours thereafter</p>

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes- Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is

**Methodology For Mark-up of NUREG-1431 Specifications
(Continued)**

"struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B
MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES CONTENTS

Mark-up:

<u>BASES</u>	<u>PAGE</u>
3.2.1	B3.2-1
3.2.2	B3.2-11
3.2.3	B3.2-18
3.2.4	B3.2-23
Methodology	(1 Page)

BASES

BACKGROUND

The purpose of the limits on the values of $F_0(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_0(Z)$ varies along the axial height (Z) of the core.

$F_0(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_0(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.7 ~~6~~, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_0(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

~~$F_0(Z)$ is measured periodically using the incore detector system. $F_0(Z)$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. The results of the three-dimensional power distribution map are analyzed to derive a measured value for $F_0(Z)$. These measurements are generally taken with the core at or near steady-state equilibrium conditions.~~

~~Using the measured three-dimensional power distributions, it is possible to derive a measured value for $F_0(Z)$. However, because this value represents a steady-state equilibrium condition, it does not include the variations in the value of $F_0(Z)$ that are present during nonequilibrium situations, such as load following.~~

~~To account for these possible variations, a transient $F_0(Z)$ is also calculated based on the steady state value of $F_0(Z)$. In this case the steady state $F_0(Z)$ is adjusted by an elevation dependent factor, $W(Z)$, that accounts for the calculated worst-case transient conditions.~~

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of

BASES

BACKGROUND
(continued)

the appropriate LCOs, including the limits on
AFD, QPTR, and control rod insertion.

APPLICABLE
SAFETY ANALYSES

This LCO's principal effect is to preclude core power distributions that violate could lead to violation of the following fuel design criteria criterion:

- a. During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1);
- b. During a the Condition 2 partial loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition;
- c. During an ejected rod accident, the average fuel pellet enthalpy at the hot spot in irradiated energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

Limits on $F_0(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the LOCA peak cladding temperature is typically most limiting.

$F_0(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_0(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_0(Z)$ satisfies Criterion 2 of 10 CFR 50.36(g)(2)(1) the NRC Policy Statement.

BASES
LCO

The Heat Flux Hot Channel Factor, F₀(Z), shall be limited by the following relationships:

$$F_0(Z) \leq \frac{GFQ}{P} K(Z) \quad \text{for } P > 0.5$$

$$F_0(Z) \leq \frac{GFQ}{0.5} K(Z) \quad \text{for } P < 0.5$$

$$F_0(Z) \leq \frac{GFQ}{P} K(Z) \quad \text{for } P > 0.5$$

$$F_0(Z) \leq \frac{GFQ}{0.5} K(Z) \quad \text{for } P \leq 0.5$$

where: $GFQ = F_0^{RTP}$ and is the F₀(Z) limit at RATED THERMAL POWER (RTP) provided in the COLR.

K(Z) is the normalized F₀(Z) normalization factor for as a function of core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

For this facility, the actual values of F_0^{RTP} , GFQ and K(Z) are given in the COLR; however, F_0^{RTP} , GFQ is normally a number on the order of [2.32] 2.45, and K(Z) is a function that looks like the one provided in Figure B 3.2.1B-1.

For Relaxed Axial Offset Control operation, F₀(Z) is approximated by F₀^c(Z) and F₀^m(Z). Thus, both F₀^c(Z) and F₀^m(Z) must meet the preceding limits on F₀(Z).

An F₀^c(Z) evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value (F₀^m(Z)) of F₀(Z). Then, the computed heat flux hot channel factor, F₀^c(Z) is obtained by the equation:

$$F_0^c(Z) = F_0^m(Z) [1.0815] (1.03) (1.05) = F_0^m(Z)$$

where [1.0815] 1.03 is a factor that accounts for fuel manufacturing tolerances and 1.05 is a factor that accounts for flux map measurement uncertainty.

F₀^c(Z) is an excellent approximation for F₀(Z) when the reactor is at the steady state power at which the incore flux map was taken.

(continued)

BASES

LCO
(continued)

The expression for $F_0^H(Z)$ is:

$$F_0^H(Z) = F_0^C(Z) W(Z)$$

where $W(Z)$ is a cycle dependent function that accounts for power distribution transients encountered during normal operation. $W(Z)$ is included in the COLR.

The $F_0(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. ~~Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_0(Z)$ limits. If $F_0(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.~~

Violating the LCO limits for $F_0(Z)$ may produce unacceptable consequences if a design basis event occurs while $F_0(Z)$ is outside its specified limits.

~~If the power distribution measurements are performed at a power level less than 100% RTP, then the $F_0^H(Z)$ and $F_0^C(Z)$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with F_0^H assures compliance with the LCO at all power levels.~~

APPLICABILITY

The $F_0(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by $\geq 1\%$ RTP for each 1% by which $F_0^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_0^C(Z)$ is $F_0^H(Z)$ multiplied by a factor ξ which accounting for manufacturing tolerances and measurement uncertainties. $F_0^H(Z)$ is the measured value of $F_0(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)

BASES

 ACTIONS
 (continued)
A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by $\approx 1\%$ for each 1% by which $F_0^c(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

Reduction in the Overpower ΔT trip setpoints by $\approx 1\%$ for each 1% by which $F_0^c(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

Verification that $F_0^c(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions. Inherent in this action is identification of the cause of the out of limit condition, and the correction of the cause, to the extent necessary to allow safe operation at the higher power level. The allowable power level is determined by extrapolating $F_0^c(Z)$. SR 3.2.1.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints.

B.1

If it is found that the maximum calculated value of $F_0(Z)$ that can occur during normal maneuvers, $F_0^m(Z)$, exceeds its specified limits, there exists a potential for $F_0^c(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD by $\approx 1\%$ for each 1% by which $F_0^m(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, restricts the axial flux

(continued)

BASES

distribution such that even if a transient occurred, core peaking factors limits are not exceeded.

ACTIONS
(continued)C.1

If Required Actions A.1 through A.4 or B.1 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during the first power ascension after a refueling during power ascensions following a plant shutdown (leaving MODE 1). The note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_0^c(Z)$ and $F_0^m(Z)$ are within their specified limits after a power rise of more than 10% 20% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_0^c(Z)$ and $F_0^m(Z)$ could not have previously been measured in this for a reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_0^c(Z)$ and $F_0^m(Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_0^c(Z)$ and $F_0^m(Z)$ following a power increase of more than 10% 20%, ensures that they are verified as soon as within 12 24 hours from when equilibrium conditions are achieved at RTP (or any other level for extended operation) → is achieved. Equilibrium conditions are achieved when the core is sufficiently stable such that the uncertainties associated with the measurement are valid. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_0^c(Z)$ and $F_0^m(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% 20% increase in power level above the last verification. It only requires verification after a power level is achieved for extended

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

operation that is 10% 20% higher than that power at which F₀ was last measured.

SR 3.2.1.1

Verification that F₀^c(Z) is within its specified limits involves increasing F₀^m(Z) to allow for manufacturing tolerance and measurement uncertainties in order to obtain F₀^c(Z). Specifically, F₀^m(Z) is the measured value of F₀(Z) obtained from incore flux map results and F₀^c(Z) = F₀^m(Z) [1.0015] (1.03) (1.05) (Ref. 4 2). F₀^c(Z) is then compared to its specified limits.

The limit with which F₀^c(Z) is compared varies inversely with power above 50% RTP and directly with a function called K(Z) provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures (and meeting the 100% RTP F₀(Z) limit) provides assurance that the F₀^c(Z) limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by ≥ 10% 20% RTP since the last determination of F₀^c(Z), another evaluation of this factor is required [12] 24 hours after achieving equilibrium conditions at this higher power level (to ensure that F₀^c(Z) values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 3.2.1.2

~~The nuclear design process includes calculations performed to determine that the core can be operated within the F₀(Z) limits.~~ Because flux maps are taken in steady state equilibrium conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called W(Z). Multiplying the measured total peaking factor, F₀^c(Z), by W(Z) gives the

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.2 (continued)

maximum $F_0(Z)$ calculated to occur in normal operation, $F_0^M(Z)$.

The limit with which $F_0^M(Z)$ is compared varies inversely with power and directly with the function $K(Z)$ provided in the COLR.

The $W(Z)$ curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_0^C(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

- a. Lower core region, from 0 to 15% inclusive; and
- b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. ~~When $F_0^M(Z)$ is evaluated and found to be within its limit measured determined,~~ an evaluation of the expression below is required to account for any increase to $F_0^M(Z)$ that may occur and cause the $F_0(Z)$ limit to be exceeded before the next required $F_0(Z)$ evaluation.

If the two most recent $F_0(Z)$ evaluations show an increase in the expression

$$\text{maximum over } z \quad \left[\frac{F_0^C(Z)}{K(Z)} \right]$$

it is required to meet the $F_0(Z)$ limit with the last $F_0^M(Z)$ increased by a factor of ~~[1.02] = 2 percent which is specified in the COLR,~~ or to evaluate $F_0(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_0(Z)$ from exceeding its limit for any significant period of time without detection.

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.2.1.2 (continued)

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP or at a reduced power at any other time, ensures and meeting the 100% RTP F₀(Z) limit, provides assurance that the F₀(Z) limit ~~is~~ will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

F₀(Z) is verified at power levels $\geq 10\%$ ~~20%~~ RTP above the THERMAL POWER of its last verification, ~~[12]~~ 24 hours after achieving equilibrium conditions to ensure that F₀(Z) is within its limit at higher power levels.

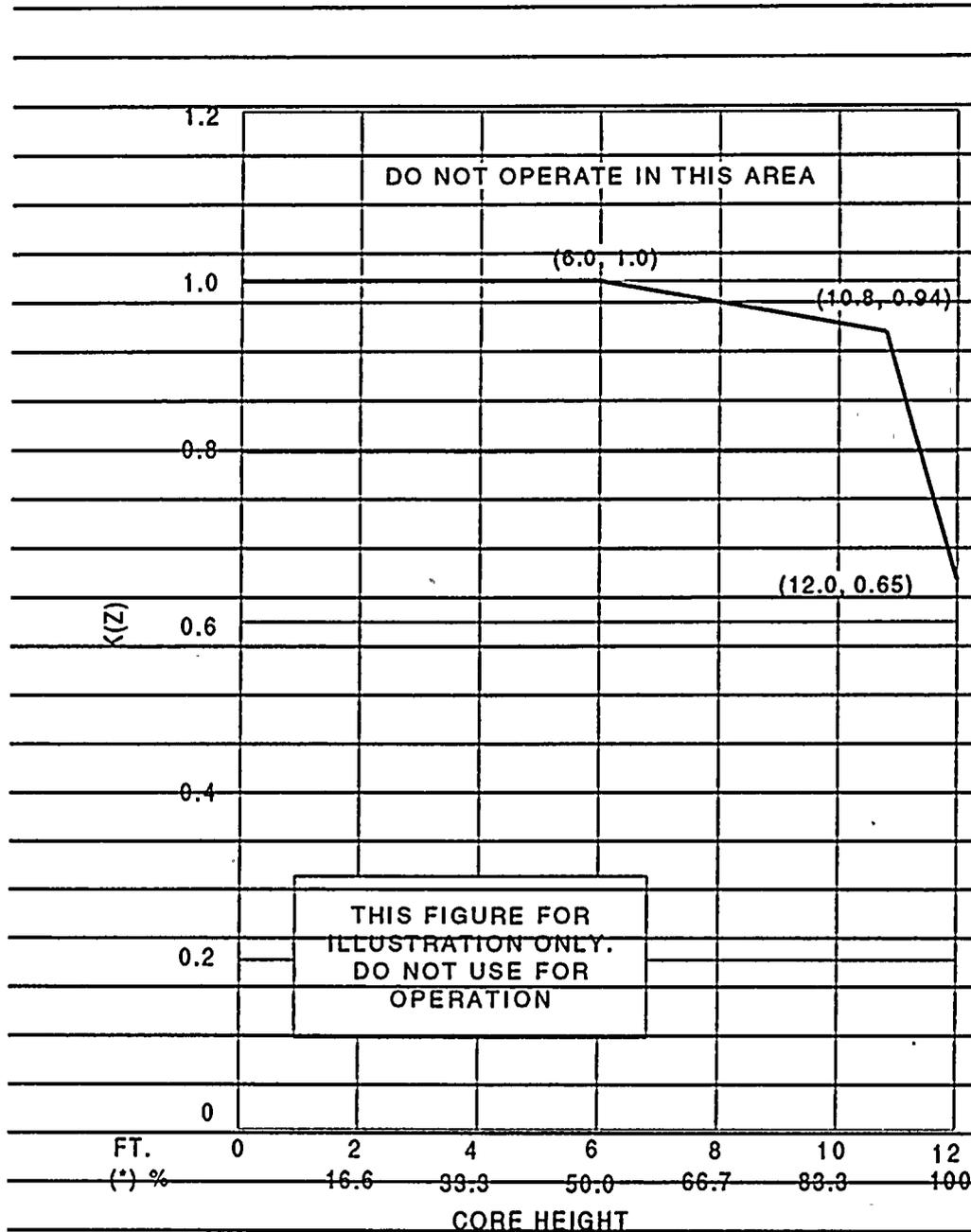
The Surveillance Frequency of 31 EFPD is ~~normally~~ adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of F₀(Z) evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46, 1974.
- ~~2. Regulatory Guide 1.77, Rev. 0, May 1974.~~
- ~~3. 10 CFR 50, Appendix A, GDC 26.~~
- 4 ~~2.~~ WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties." June 1988.

(continued)



*For core height of 12 feet

B 3.2 POWER DISTRIBUTION LIMITS Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized F₉(Z) as a Function of Core Height

(continued)

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident operational transients and any transient condition arising from events of moderate frequency analyzed in the safety analyses.

$F_{\Delta H}^N$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F_{\Delta H}^N$ is a measure of the maximum total power produced in a fuel rod. $F_{\Delta H}^N$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F_{\Delta H}^N$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F_{\Delta H}^N$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F_{\Delta H}^N$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables. Compliance with these LCOs along with the LCOs governing shutdown and control rod insertion and alignment, maintains the core limits on power distribution on a continuous basis.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to [1.3] using the [W3] CHF correlation. All DNB limited transient events are assumed to begin with an $F_{\Delta H}^N$ value that satisfies the LCO requirements.

(continued)

BASES

BACKGROUND
(continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE
SAFETY ANALYSES

Limits on F_{AH}^N preclude core power distributions that exceed the following fuel design limits:

- a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;
- b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;
- c. During an ejected rod accident, the energy deposition to the average fuel pellet enthalpy at the hot spot in irradiated fuel must not exceed 280 cal/gm [Ref. 1]; and
- d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

~~For transients that may be DNB limited, the Reactor Coolant System flow and F_{AH}^N are the core parameters of most importance. The limits on F_{AH}^N ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion of [1.3] using the [W3] CHF applicable to a specific DNBR correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB condition.~~

The allowable F_{AH}^N limit increases with decreasing power level. This functionality in relationship between power and F_{AH}^N is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

this variable value of F^N_{AH} in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with an limiting initial F^N_{AH} as a function of power level defined by the COLR F^N_{AH} limit equation in the COLR.

The LOCA safety analysis indirectly models also uses F^N_{AH} as an input parameter. The Nuclear Heat Flux Hot Channel Factor (F₀(Z)) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature [Ref. 3].

The fuel is protected in part by compliance with Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this:
LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1.7 6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor (F^N_{AH})," and LCO 3.2.1, "Heat Flux Hot Channel Factor (F₀(Z))."

F^N_{AH} and F₀(Z) are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

F^N_{AH} satisfies Criterion 2 of the NRC Policy Statement 10 CFR 50.36(c)(2)(ii).

LCO

F^N_{AH} shall be maintained within the limits of the relationship provided in the COLR.

The F^N_{AH} limit identifies is representative of the coolant flow channel with the maximum enthalpy rise. This channel has the least heat removal capability and thus the highest probability for a DNB condition.

The limiting value of F^N_{AH}, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional margin allowance for higher radial peaking factors from reduced

(continued)

BASES

LCO
(continued)

thermal feedback and greater control rod insertion at low power levels. The limiting value of F^N is allowed to increase 0.3% by a cycle dependent factor, PF_{LD}, specified in the COER for every a 1% RTP reduction in THERMAL POWER.

If the power distribution measurements are performed at a power level less than 100% RTP then the F^N values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with F^N assures compliance with the LCO at all power levels.

APPLICABILITY

The F^N limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power. Specifically, the design bases events that are sensitive to F^N in other modes (MODES 2 through 5) have significant margin to DNB, and therefore, there is no need to restrict F^N in these modes.

ACTIONS

A.1.1

With F^N exceeding its limit, the unit is allowed 4 hours to restore F^N to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring F^N within its power dependent limit. When the F^N limit is exceeded, the DNBR limit is not likely to be violated in steady state operation, because events that could significantly perturb the F^N value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore F^N to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time. The restoration of the peaking factor to within its limits by power reduction or control rod movement does not restore compliance with the LCO. Thus, this condition can not be exited until a valid surveillance demonstrates compliance with the LCO.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of F^N within 24 hours in accordance with SR 3.2.2.1.

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of F^N must be done prior to exceeding 50% RTP, prior to exceeding

(continued)

BASES

ACTIONS

A.1.1 (continued)

75% RTP, and within 24 hours after reaching or exceeding 95% RTP; however, THERMAL POWER does not have to be reduced to comply with these requirements. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of F^N is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux-High to \leq 55% RTP in accordance with Required Action A.1.2.2. Reducing RTP THERMAL POWER to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints; however, for extended operations at the reduced power level, the reduced trip setpoints are required to protect against involving positive reactivity excursions. This is a sensitive operation that may inadvertently trip-actuate the Reactor Protection System.

A.2

Once actions have been taken to restore F^N to within its limits per Required Action A.1.1, or the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of F^N verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which

(continued)

BASES

ACTIONS

A.2 (continued)

is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate F^N_{AH}.

A.3

Verification that F^N_{AH} is within its specified limits after an out of limit occurrence ensures that the cause that led to the F^N_{AH} exceeding its the F^N_{AH} limit is identified to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the F^N_{AH} limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP. SR 3.2.2.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints. When F^N_{AH} is measured at reduced power levels, the allowable power level is determined by evaluating F^N_{AH} for higher power levels.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

SR 3.2.2.1 is modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving MODE 1). The Note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. Equilibrium conditions are achieved when the core is sufficiently stable such that uncertainties associated with the measurement are valid.

The value of F^N_{AH} is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of F^N_{AH} from the measured flux distributions. The measured value of F^N_{AH} must be multiplied by 1.04 to account for

(continued)

SURVEILLANCE
REQUIREMENTS

SR 3.2.2.1 (continued)

measurement uncertainty before making comparisons to the F_{NH} limit.

After each refueling, F_{NH} must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that F_{NH} limits are met at the beginning of each fuel cycle. Performing this Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and meeting the 100% RTP F_{NH} limit, provides assurance that the F_{NH} limit is met when RTP is achieved because peaking factors generally decrease as power level is increased.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the F_{NH} limit cannot be exceeded for any significant period of operation.

REFERENCES

1. Regulatory Guide 1.77, Rev. [0] 0, May 1974.
 2. 10 CFR 50, Appendix A, GDC 26.
 3. 10 CFR 50.46.
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(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3B AXIAL FLUX DIFFERENCE (AFD) (~~Relaxed Axial Offset Control (RAOC) Methodology~~)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

~~Relaxed Axial Offset Control (RAOC) is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.~~

~~The AFD is monitored on an automatic basis using the unit plant process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.~~

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, ~~Constant Axial Offset Control (CAOC)~~, is used to control axial power distribution in day to day operation (Ref. 1). ~~CAOC~~ This requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

~~The CAOC This operating space is typically smaller and lies within the RAOC operating space. Control within the CAOC This operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.~~

(continued)

BASES (continued)

APPLICABLE

The AFD is a measure of the axial power distribution skewing SAFETY ANALYSES to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentatively wide AFD limits. One dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_0(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition 2, 3, or 4 II, III, or IV events. This compliance with these limits ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition 4 IV event is the LOCA. The most important Condition 3 III event is the complete loss of forced RCS flow accident. The most important Condition 2 II events are uncontrolled bank withdrawal and boration or dilution accidents. Condition 2 II accidents simulated to begin from within the AFD limits are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of the NRC Policy Statement 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion

(continued)

BASES

LCO
(continued)

of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from power level turbine load changes.

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom power range detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as $\% \Delta$ flux or $\% \Delta I$.

The AFD limits are provided in the COLR. -
~~Figure B 3.2.3B 1 shows typical RAOC AFD limits.~~ The AFD limits for RAOC do not depend on the target flux difference. However, the target flux difference AFD may be used to minimize changes in the axial power distribution.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 T1, T2, or IV event occurs while the AFD is outside its specified limits.

 APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

 ACTIONS

A.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of

(continued)

BASES

ACTIONSA.1 (continued)

30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.2.3.1

~~The AFD is monitored on an automatic basis using the unit process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels is outside its specified limits.~~

This Surveillance verifies that the AFD, as indicated by the ~~each OPERABLE~~ NIS excore channel, is within its specified limits, and is consistent with the status of the AFD monitor alarm. ~~With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.~~

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.
 2. ~~R. W. Miller et al., "Relaxation of Constant Axial Offset Control: F₀ Surveillance Technical Specification," WCAP-10217(NP), June 1983. WCAP-10216-P-A, Revision 1A, Relaxation of Constant Axial Offset Control, F₀ Surveillance Technical Specification, February 1994 (Westinghouse Proprietary)~~
 3. FSAR, Chapter ~~[15]~~ 15.
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(continued)

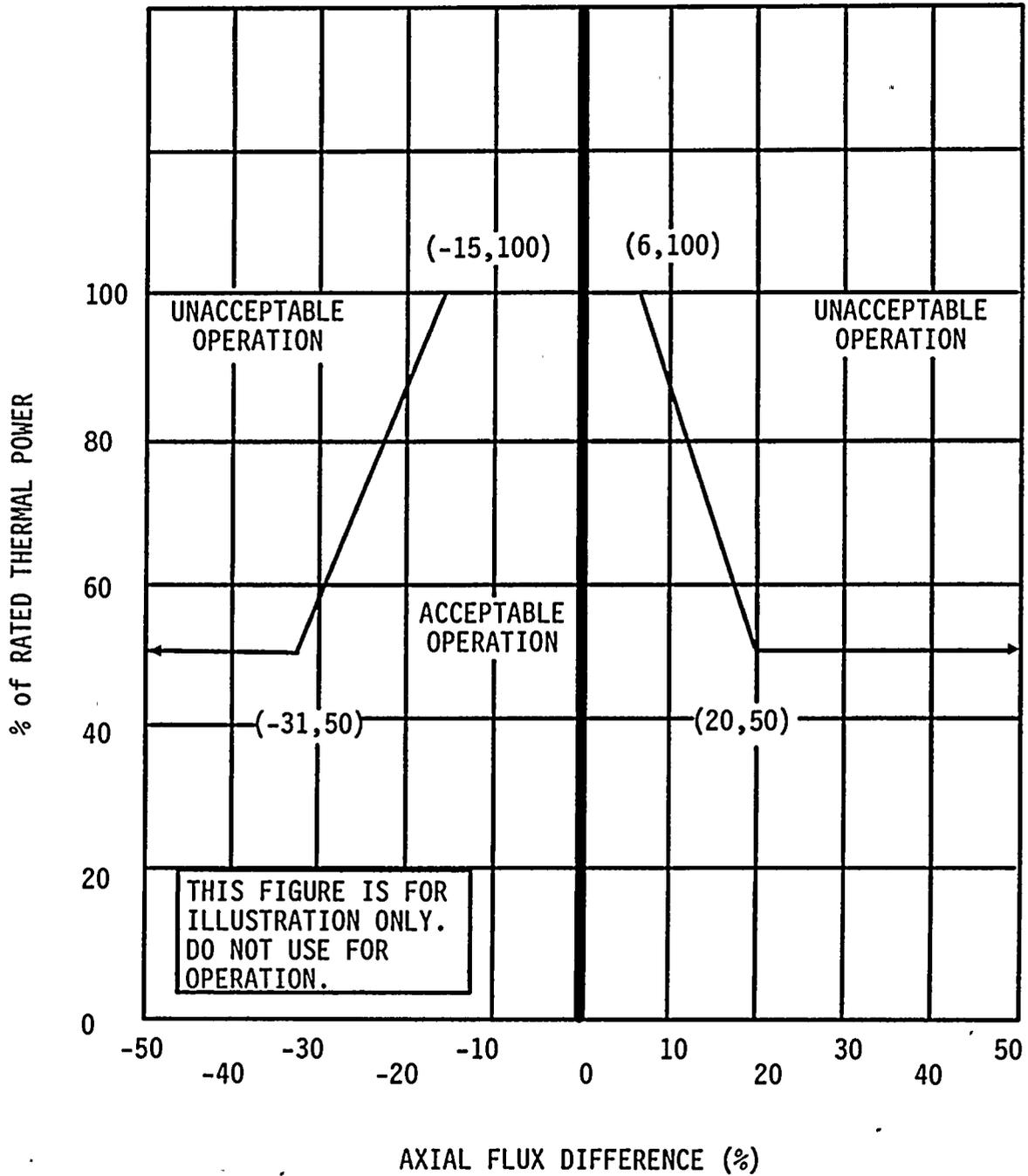


Figure B 3.2.3B-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.7.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE
SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

- a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);
- b. During a ~~the Condition II partial~~ loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;
- c. During an ejected rod accident, ~~the energy deposition to the average fuel pellet enthalpy at the hot spot in irradiated fuel~~ must not exceed 280 cal/gm (Ref. 2); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_0(Z)$), the Nuclear Enthalpy Rise Hot

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

(continued)

Channel Factor (F_{ch}^N), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that F_{ch}^N and $F_0(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the F_{ch}^N and $F_0(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of the NRC Policy Statement ~~10 CFR 50.36(c)(2)(ii)~~.

LCO

The QPTR limit of 1.02, at above which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_0(Z)$ and (F_{ch}^N) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 \leq 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the F_{ch}^N and $F_0(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient

(continued)

BASES

ACTIONS

A.1 (continued)

time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

A.2

After completion of Required Action A.1, the QPTR ~~alarm may still be in its alarmed state may still exceed its limits.~~ As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors F_{AH}^N and $F_0(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on F_{AH}^N and $F_0(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate F_{AH}^N and $F_0(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although F_{AH}^N and $F_0(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded

(continued)

BASES

ACTIONS

A.4 (continued)

and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Actions A.1 and A.2, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded remains above the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to eliminate the indicated tilt recalibrated to show a zero QPTR prior to increasing THERMAL POWER to above the limit of Required Actions A.1 and A.2. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by a Note that states that the indicated tilt is not eliminated QPTR is not zeroed out until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). This Note is intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the excore detectors are normalized to eliminate the indicated tilt flux tilt is zeroed out (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_0(Z)$ and F_{AH}^N are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the peaking factor verification cannot be performed within 24 hours due to the non-equilibrium core conditions, a maximum time of 48 hours is allowed for the completion of the verification.

(continued)

BASES

ACTIONS

A.6 (continued)

~~core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Actions A.1 and A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.~~

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after must be completed when the excore detectors have been normalized to eliminate the indicated tilt calibrated to show zero tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after are only required if the excore detectors are calibrated were normalized to show zero tilt and the core returned to power per Required Action A.5.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to $\leq 50\%$ RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is $\leq 75\%$ RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1, if more than one input from Power Range Neutron Flux channels are inoperable.

~~Input from a Power Range Neutron Flux channel is considered to be OPERABLE if the upper and lower detector currents are obtainable. The remaining portion of the channel (the electronics required to provide the channel input to the QPTR alarm) need not be OPERABLE.~~

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.1 (continued)

within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room. ~~When the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.~~

~~When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.~~

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after only when the input from one or more Power Range Neutron Flux channels are is inoperable and the THERMAL POWER is \approx 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one or more power range channels is are inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 for three and four loop cores.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.2 (continued)

core flux map, to generate an incore QPTR. Therefore, incore QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.
2. Regulatory Guide 1.77. Rev ~~01~~. May 1974.
3. 10 CFR 50, Appendix A, GDC 26.

Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is

Methodology For Mark-up of NUREG-1431 Bases

"struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases, as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(2 Pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.2

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>JUSTIFICATION</u>
3.2-01	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 6B).
3.2-02	Consistent with the CTS, retain the requirement for performing F_0 after a 20 percent change in power (versus the 10 percent value specific in the ITS).
3.2-03	Consistent with Traveler TSTF-98, Rev. 1, the factor by which the F_0 must be adjusted on increasing F_0 measurements is moved to the COLR. This change is acceptable because the factor is normally contained in the COLR, and it removes detail not required to be contained in TS.
3.2-04	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 6B).
3.2-05	In accordance with Traveler TSTF-25, clarify that the excore detectors are normalized to indicate zero tilt. This is only a clarification of NUREG-1431 wording and is acceptable.
3.2-06	Consistent with Traveler TSTF-95, the time allowed for resetting the power range neutron flux - high setpoint if F_0 or $F_{\Delta H}^N$ is outside their limits is extended from 8 hours to 72 hours. As written, the Completion Time of 8 hours to reduce the power range neutron flux-high Trip Setpoints presents an unjustified burden on the operation of the plant. A Completion Time of 72 hours will allow time to perform a second flux map to confirm the results, or determine that the Condition was temporary, without implementing an unnecessary Trip Setpoint change, during which there is increased potential for a plant transient and human error. Following a significant power reduction, at least 24 hours are required to reestablish steady state xenon prior to taking a flux map, and approximately 8 to 12 hours to obtain a flux map, and analyze the data. A significant potential for human error can be created through requiring the Trip Setpoints to be reduced within the same time frame that a unit power reduction is taking place, and within the current 8 hour period. Setpoint adjustment is estimated to take approximately, 4 hours per channel (review of plant Condition supportive of removing channels from service, tripping of bistables, setpoint adjustments, and channel restoration), adding 2 hours for necessary initial preparations (procedure preps, calibration equipment checks, obtaining tools, and approvals), it is reasonable to expect a total of 18 hours. Further, setpoint changes should only be required for extended operation in this Condition. Finally, the Bases for making this setpoint change is the same as the NUREG Bases provided for the 72 hour Completion Time of LCO 3.2.1 Required Action A.4, which is also a setpoint reduction. In summary, this change is acceptable because it would permit time to perform required flux mapping, permit orderly resetting of the high flux Trip Setpoints, and reduce the chances of an inadvertent reactor trip during the required power reduction.
3.2-07	Consistent with Traveler TSTF-97; the Note in SR 3.2.1.2 is revised by removing the phrase "is within limits and" to clarify that the actions to be taken if $F_0^c(Z)$ is increasing are required regardless of whether $F_0^c(Z)$ is within its limits.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.2

- 3.2-08 Consistent with Traveler TSTF-99, the LCO 3.2.1 (F_o Methodology), Required ACTION B.1. Completion Time for the reduction of the AFD limits if $F_o^w(Z)$ is not within limits is increased from 2 hours to 4 hours. This makes it consistent with the Completion Time associated with Required ACTION A.2. of LCO 3.2.1 (F_{xy} methodology). The change is acceptable because it eliminates an inconsistency in the ITS.
- 3.2-09 For consistency with CTS 3.2.4 and ITS 3.3.1, Condition D, the breakpoints for the Applicability of the surveillances in the notes in ITS SR 3.2.4.1 and SR 3.2.4.2 are modified to be applicable at less than or equal to 75 percent RTP, and greater than 75 percent RTP, respectively. This is an administrative change that retains CTS requirements.
- 3.2-10 Consistent with Traveler TSTF-110, this change moves requirements for increased surveillance frequencies in the event of inoperable alarms to licensee controlled documents. This change is acceptable because it removes requirements regarding alarms and alarm responses that are not necessary to be in the TS to protect public health and safety.
- 3.2-11 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.2-12 Consistent with CTS, the required time for completion of a flux map for determination of the heat flux hot channel factor is changed from 12 hours to 24 hours after achieving equilibrium Conditions. The proposed change affects SR 3.2.1.1 and SR 3.2.2.1. The proposed time (24 hours) is a reasonable time period for the completion of the surveillance and does not allow for plant operation in an uncertain condition for a protracted time period. This change is consistent with the TS requirements of Specification 3.0.4 (and associated Bases) that allow 24 hours for the completion of a surveillance after prerequisite plant conditions are attained and for which an exception to Specification 4.0.4 was provided.
- 3.2-13 This change retains the CTS for the performance of peaking factor determinations following plant shutdowns. The CTS, through the exemption to Specification 4.0.4, allows prerequisite plant conditions to be obtained prior to requiring that the surveillance be completed.
- 3.2-14 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.2-15 This change incorporates Traveler TSTF-109. ACTION A.2. would require the QPTR be determined rather than performing a specific surveillance because more than one surveillance can be used to determine QPTR. SR 3.2.4.1 was revised to retain allowance that SR 3.2.4.2 may be performed in lieu of SR 3.2.4.1. The note for SR 3.2.4.2 is changed to require performance if one "or more" QPTR inputs are inoperable with RTP > 75 percent. These changes are acceptable because they clarify the ITS regarding frequency and use of incore flux monitoring for QPTR measurement. The changes reflect that incore detectors provide an acceptable QPTR determination during all plant Conditions.
- 3.2-16 This change would require both transient and static F_o measurements be determined when performed for Required ACTIONS 3.2.4 A.3 and A.6. The intent of the Required ACTIONS is to verify that $F_o(Z)$ is within its limit. $F_o(Z)$ is approximated by $F_o^c(Z)$ (which is obtained via SR 3.2.1.1) and $F_o^w(Z)$ (which is obtained via SR 3.2.1.2). Thus, both $F_o^c(Z)$ and $F_o^w(Z)$ must be established to verify $F_o(Z)$. This change is consistent with Traveler WOG-105.
- 3.2-17 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.2-18 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.2-19 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

CONVERSION COMPARISON TABLE

(3 PAGES)

NUMBER	TECH SPEC CHANGE DESCRIPTION	APPLICABILITY			
		DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.2-01	Substitute overpower CPSES N-16 for overpower ΔT reactor trip function.	No	Yes	No	No
3.2-02	Retain CTS requirement for performing F_o after a 20% change in power (versus the 10% value specific in the ISTS).	Yes	Yes	No, CTS specifies 10%.	No, CTS specifies 10%.
3.2-03	Consistent with TSTF-98, Rev. 1, the factor by which the F_o must be adjusted on increasing FQ measurements is moved to the COLR.	Yes	Yes	Yes	Yes
3.2-04	SR is applicable to plants using the Westinghouse Constant Axial Offset Control methodology. TU Electric uses the methodology described in RXE-90-006-P-A, described in CTS 6.9.1.6b, which ties the target flux difference surveillance frequency to the frequency at which the $F_o^{w(Z)}$ peaking factor is verified.	No	Yes	No	No
3.2-05	Per a proposed revision to WOG-95, this change clarifies that the excore detectors are normalized to indicate zero tilt.	Yes	Yes	Yes	Yes
3.2-06	Consistent with TSTF-95, increases time to reset Hi Flux setpoints from 8 to 72 hours.	Yes	Yes	Yes	Yes
3.2-07	Consistent with TSTF-97, clarifies the ACTIONS for increasing $F_o^{w(Z)}$.	Yes	Yes	Yes	Yes
3.2-08	Consistent with TSTF-99, increases the allowance for AFD restrictions from 2 hours to 4 hours.	Yes	Yes	Yes	Yes
3.2-09	The breakpoints for the Applicability of the surveillances in the notes in ITS SR 3.2.4.1 and SR 3.2.4.2 are modified to be applicable at less than or equal to 75% RTP and greater than 75% RTP, respectively.	Yes	Yes	Yes	Yes
3.2-10	Consistent with TSTF-110, moves requirements for increased surveillance frequencies in the event of inoperable alarms to licensee controlled documents.	Yes, to ECG and FSAR.	Yes, to TRM.	Yes, to FSAR.	Yes, to FSAR.

NUMBER	TECH SPEC CHANGE DESCRIPTION	APPLICABILITY			
		DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.2-11	This change rearranges the notes applicable to LCO 3.2.3. AFD (Constant Axial Offset Control plants only, CPSES specific) from three separate notes affecting different portions of the LCO into a common note block. This change is consistent with TSTF-164.	No	Yes	No	No
3.2-12	Consistent with CTS, the required time for completion of a flux map for determination of the heat flux hot channel factor is changed from 12 hours to 24 hours after achieving equilibrium Conditions. The proposed change affects SR 3.2.1.1 and SR 3.2.2.1.	Yes	Yes	No, see CN 3.2-17.	No, see CN 3.2-17.
3.2-13	This change retains the CTS for the performance of peaking factor determinations following plant shutdowns.	Yes	Yes	Yes	Yes
3.2-14	This change retains the Wolf Creek current licensing basis (License Amendment 61) for Required ACTION 3.2.2 A.2.	No	No	Yes	No
3.2-15	This change partially incorporates the industry traveler TSTF-109. ACTION A.2 now requires the QPTR be determined rather than performing a specific surveillance because more than 1 surveillance can be used to determine QPTR. The note for SR 3.2.4.2 is changed to require performance if one "or more" QPTR inputs are inoperable.	Yes	Yes	Yes	Yes
3.2-16	Requires both transient and static F_0 measurements when performed for Required ACTIONS 3.2.4 A.3 and A.6.	Yes	Yes	Yes	Yes
3.2-17	The frequency requirement for performing F_0 measurements has been revised to conform to the CTS which did not specify a Completion Time.	No, see CN 3.2-12.	No, see CN 3.2-12.	Yes	Yes
3.2-18	This change modifies the QPTR requirements in NUREG-1431, Rev. 1, for Wolf Creek, to retain some CTS requirements and to incorporate revisions to Required ACTIONS proposed by Traveler WOG-95.	No	No	Yes	No

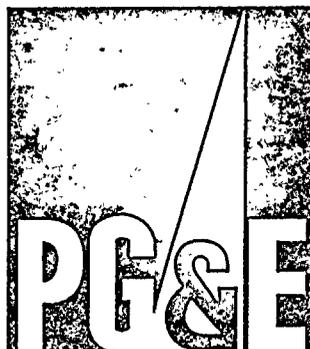
TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.2-19	This Wolf Creek specific change modifies SR 3.2.3.1 to resolve a literal compliance concern regarding the use of the LCO note to modify the SR.	No	No	Yes	No

JLS Conversion to
Improved Technical Specifications
Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706236042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 3/4.3 - Instrumentation

ITS 3.3 - Instrumentation



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.3

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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	25 pages
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METHODOLOGY	3 pages

CROSS-REFERENCE TABLE FOR 3/4.3
Sorted By Current TS

<u>Current Technical Specifications</u>				<u>Improved Technical Specifications</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.3.1	LCO		01-35-LG	3.3.1	LCO		
3.3.1	APPLIC.	TBL 3.3-1		3.3.1	APPLIC.	TBL3.3.1-1	
3.3.1	ACTIONS	TBL 3.3-1	01-01-A	3.3.1	ACTIONS	A thru X	
3.3.1	New	*Note	01-01-A	3.3.1	ACTIONS	NOTE	
3.3.1	SR	4.3.1.1		3.3.1	SRS	Various	
3.3.1	SR	4.3.1.2	01-02-LG	3.3.1	SR 3.3.1.16		
3.3.1	SR	4.3.1.2	01-03-LS1	3.3.1	SR 3.3.1.16		
3.3.1	TABLE	3.3-1		3.3.1	TABLE	3.3.1-1	
3.3.1	TABLE	3.3-2	01-35-LG	BASES			
3.3.1	TBL 3.3-2	NOTE (1)		3.3.1	SR3.3.1.16	NOTE	
3.3.1	TBL 3.3-2	NOTE (2)		FSAR			
3.3.1	TABLE	4.3-1		3.3.1	TABLE	3.3.1-1	
TABLE 3.3-1& 4.3-1(ordered by function, not Table)				TABLE 3.3-1, ACTIONS & SRS			
F-Unit 1	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 1	
F-Unit 1	ACTION	1		3.3.1	ACTION B		3.3-106
F-Unit 1	ACTION	11		3.3.1	ACTION C		3.3-122
F-Unit 1	SR	TADOT-R	01-32-LG	3.3.1	SR 3.3.1.14		
F-Unit 2.a	APPLIC.			TBL 3.3.1-1	APPLIC.	FUNC. 2.a	
F-Unit 2.a	ACTION	2		3.3.1	ACTION D		
F-Unit 2.a	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit 2.a	SR	Ch Cal-D	01-21-A	3.3.1	SR 3.3.1.2		
F-Unit 2.a	SR	COT-Q		3.3.1	SR 3.3.1.7		
F-Unit 2.a	SR	Ch Cal-M	01-21-A	3.3.1	SR 3.3.1.3		
F-Unit 2.a	SR	Ch Cal-Q	01-21-A	3.3.1	SR 3.3.1.6		
F-Unit 2.a	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.11		
F-Unit 2.b	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 2.b	
F-Unit 2.b	ACTION	2.1(new)	01-06-LS2	3.3.1	ACTION E		

CROSS-REFERENCE TABLE FOR 3/4.3
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<u>Current Technical Specifications</u>				<u>Improved Technical Specifications</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
F-Unit 2.b	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit 2.b	SR	Ch Cal-R	01-39-A	3.3.1	SR 3.3.1.11		
F-Unit 2.b	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.11		
F-Unit 2.b	SR	COT-S/U	01-22-M	3.3.1	SR 3.3.1.8		3.3-49
F-Unit 2.b	SR	COT-Q	01-22-M	3.3.1	SR 3.3.1.8		3.3-49
F-Unit 3	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 3.a	
F-Unit 3	ACTION	2.1(new)	01-06-LS2	3.3.1	ACTION E		
F-Unit 3	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.11		
F-Unit 3	SR	Ch Cal-R	01-39-A	3.3.1	SR 3.3.1.11	NOTE	
F-Unit 3	SR	COT-Q		3.3.1	SR 3.3.1.8		3.3-49
F-Unit 4	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 3.b	
F-Unit 4	ACTION	2.1(new)	01-06-LS2	3.3.1	ACTION E		
F-Unit 4	SR	Ch Cal-R	01-39-A	3.3.1	SR 3.3.1.11	NOTE	
F-Unit 4	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.11		
F-Unit 4	SR	COT-Q		3.3.1	SR 3.3.1.8		3.3-49
F-Unit 5	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 4	
F-Unit 5	ACTION	3		3.3.1	ACTION F		3.3-95
F-Unit 5	ACTION	3.1(new)	01-07-LS3	3.3.1	ACTION G		3.3-95
F-Unit 5	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit 5	SR	Ch Cal-R		3.3.1	SR 3.3.1.11		
F-Unit 5	SR	COT-S/U	01-22-M	3.3.1	SR 3.3.1.8		3.3-49
F-Unit 5	SR	COT-Q	01-22-M	3.3.1	SR 3.3.1.8		3.3-49
F-Unit6a-c	APPLIC.		01-47-A	TBL3.3.1-1	APPLIC.	FUNC. 5	
F-Unit 6.a	ACTION	4		3.3.1	ACTION I		
F-Unit 6.a	ACTION	4.1(new)	01-08-M	3.3.1	ACTION J		
F-Unit 6.b	ACTION	11	01-08-M	3.3.1	ACTION K		3.3-122
F-Unit 6.b	ACTION	4.1(new)	01-08-M	3.3.1	ACTION J		

CROSS-REFERENCE TABLE FOR 3/4.3
Sorted By Current TS

<u>Current Technical Specifications</u>				<u>Improved Technical Specifications</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
F-Unit 6.c	ACTION	5		3.3.1	ACTION L		3.3-123
F-Unit6a-c	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit6a-c	SR	Ch Cal-R	01-26-LG	3.3.1	SR 3.3.1.11		
F-Unit6a-c	SR	COT-S/U	01-28-A	3.3.1	SR 3.3.1.8		3.3-49
F-Unit6a,c	SR	COT-Q		3.3.1	SR 3.3.1.8		3.3-49
F-Unit 6 b	SR	COT-Q	01-22-M	3.3.1	SR 3.3.1.7		
F-Unit 7	APPLIC.			TBL 3.3.-1	APPLIC.	FUNC. 6	
F-Unit 7	ACTION	2.1(new)	01-45-M	3.3.1	ACTION E		
F-Unit 7	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit 7	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		
F-Unit 7	SR	Ch Cal-M	01-21-A	3.3.1	SR 3.3.1.3		
F-Unit 7	SR	Ch Cal-Q	01-21-A	3.3.1	SR 3.3.1.6		
F-Unit 7	SR	COT-Q		3.3.1	SR 3.3.1.7		
F-Unit 8	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 7	
F-Unit 8	ACTION	2.1(new)	01-45-M	3.3.1	ACTION E		
F-Unit 8	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit 8	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		
F-Unit 8	SR	COT-Q		3.3.1	SR 3.3.1.7		
F-Unit 9	APPLIC.		01-19-LS8	TBL3.3.1-1	APPLIC.	FUNC. 8.a	
F-Unit 9	ACTION	6		3.3.1	ACTION M		
F-Unit 9	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit 9	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		
F-Unit 9	SR	COT-Q		3.3.1	SR 3.3.1.7		
F-Unit 10	APPLIC.	TBL3.3-1		TBL3.3.1-1	APPLIC.	FUNC. 8.b	
F-Unit 10	ACTION	2.1(new)	01-45-M	3.3.1	ACTION E		
F-Unit 10	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit 10	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		

CROSS-REFERENCE TABLE FOR 3/4.3
Sorted By Current TS

<u>Current Technical Specifications</u>				<u>Improved Technical Specifications</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
F-Unit 10	SR	COT-Q		3.3.1	SR 3.3.1.7		
F-Unit 11	APPLIC.		01-19-LS8	TBL3.3.1-1	APPLIC.	FUNC. 9	
F-Unit 11	ACTION	6		3.3.1	ACTION M		
F-Unit 11	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit 11	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		
F-Unit 11	SR	COT-Q		3.3.1	SR 3.3.1.7		
F-Unit 12a/b	APPLIC.		01-57-LG	TBL3.3.1-1	APPLIC.	FUNC. 10	
F-Unit 12a/b	APPLIC.		01-19-LS8	TBL3.3.1-1	APPLIC.	FUNC. 10	
F-Unit 12a/b	ACTION	6		3.3.1	ACTION E		
F-Unit 12	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit 12	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		
F-Unit 12	SR	COT-Q		3.3.1	SR 3.3.1.7		
F-Unit 13a	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 14	
F-Unit 13a	ACTION	2.1(new)	01-45-M	3.3.1	ACTION E		
F-Unit 13a	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
F-Unit 13a	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		
F-Unit 13a	SR	COT-Q		3.3.1	SR 3.3.1.7		
F-Unit 13b	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC.14ab	3.3-46
F-Unit 13b	ACTION	27		3.3.1	ACTION X		3.3-46
F-Unit 13b	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		3.3-46
F-Unit 13b	SR	COT-Q		3.3.1	SR 3.3.1.7		3.3-46
F-Unit 15	APPLIC.		01-19-LS8	TBL3.3.1-1	APPLIC.	FUNC. 12	
F-Unit 15	ACTION	6	01-50-A	3.3.1	ACTION M		
F-Unit 15	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		
F-Unit 15	SR	TADOT-Q	01-16-LS40	3.3.1	SR 3.3.1.9		
F-Unit 16	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 12	
F-Unit 16	ACTION	6	01-50-A	3.3.1	ACTION M		

CROSS-REFERENCE TABLE FOR 3/4.3
Sorted By Current TS

<u>Current Technical Specifications</u>				<u>Improved Technical Specifications</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
F-Unit 16	ACTION	6	01-19-LS8	3.3.1	ACTION M		
F-Unit 16	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		
F-Unit 16	SR	TADOT-Q		3.3.1	SR 3.3.1.9		
F-Unit 17a	APPLIC.		01-48-LS4	TBL3.3.1-1	APPLIC.	FUNC. 16a	
F-Unit 17b	APPLIC.		01-48-LS4	TBL3.3.1-1	APPLIC.	FUNC. 16b	
F-Unit 17a/b	ACTION	7		3.3.1	ACTION P		
F-Unit 17a/b	SR	Ch Cal-R	01-36-M	3.3.1	SR 3.3.1.10		
F-Unit 17a/b	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.10		
F-Unit 17a/b	SR	TADOTS/U	01-24-LS9	3.3.1	SR 3.3.1.15		
F-Unit 18	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 17	
F-Unit 18	ACTION	26		3.3.1	ACTION Q		
F-Unit 18	SR	TADOT-R		3.3.1	SR 3.3.1.14		
F-Unit 19	APPLIC.		01-49-LS18	TBL3.3.1-1	APPLIC.	FUNC. 17	
F-Unit 19	ACTION	6	01-49-LS18	3.3.1	ACTION M		
F-Unit 19	SR	TADOT-R		3.3.1	SR 3.3.1.14		
Functional Units 20, 21, 22, and 24 are not consistently numbered in the CTS between Table 3.3-1 and 4.3-1, however their nomenclature is consistent as is the APPLICABILITY, ACTIONS and Surveillances. The Functional Unit grouping that follows is per CTS Table 3.3-1 Functional Unit description and number except for Functional Unit 24 that is only shown on Table 4.3-1. Those Functional Units with mixed numbers show both, and the * Functional Units are from Table 4.3-1.							
Reactor Trip Breakers							
F-Unit 20	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 19	
F-Unit 20	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 19	
F-Unit 20	ACTION	10	01-14-A	3.3.1	ACTION R		
F-Unit 20	ACTION	11		3.3.1	ACTION C		3.3-122
F-Unit 21*	SR	TADOT-M	01-32-LG	3.3.1	SR 3.3.1.4		
(New) Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms							
F-Unit(new)	APPLIC.		01-14-A	TBL3.3.1-1	APPLIC.	FUNC. 20	
F-Unit(new)	ACTION	12	01-14-A	3.3.1	ACTION U		3.3-106

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F-Unit(new)	ACTION	11	01-14-A	3.3.1	ACTION C		3.3-122
F-Unit(new)	SR	TADOT-M	01-14-A	3.3.1	SR 3.3.1.4		
Automatic Trip and Interlock Logic							
F-Unit 21	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 21	
F-Unit 21	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 21	
F-Unit 21	ACTION	26		3.3.1	ACTION Q		
F-Unit 21	ACTION	11		3.3.1	ACTION C		3.3-122
F-Unit 22*	SR	ACT-M		3.3.1	SR 3.3.1.5		
Reactor Trip System Interlock- Intermediate Range Neutron Flux, P-6							
F-Unit 22a	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 18a	
F-Unit 22a	ACTION	8		3.3.1	ACTION S		3.3-44
F-Unit 20a*	SR	Ch Cal-R		3.3.1	SR 3.3.1.11		
F-Unit 20a*	SR	COT-R		3.3.1	SR 3.3.1.13		
Reactor Trip System Interlock- Low Power Reactor Trips Block, P-7							
F-Unit 22b	APPLIC.		01-51-LG	TBL3.3.1-1	APPLIC.	FUNC. 18b	
F-Unit 22b	ACTION	8.1	01-12-M	3.3.1	ACTION T		3.3-44
F-Unit 22b	ACTION	8.1	01-05-A	3.3.1	ACTION T		3.3-44
F-Unit 20b*	SR	Ch Cal-R	01-51-LG	3.3.1	SR 3.3.1.11		
F-Unit 20b*	SR	COT-R	01-51-LG	3.3.1	SR 3.3.1.13		
Reactor Trip System Interlock- Power Range Neutron Flux, P-8							
F-Unit 22c	APPLIC.		01-37-A	TBL3.3.1-1	APPLIC.	FUNC. 18c	
F-Unit 22c	ACTION	8.1(new)	01-12-M	3.3.1	ACTION T		3.3-44
F-Unit 22b	ACTION	8.1(new)	01-05-A	3.3.1	ACTION S		3.3-44
F-Unit 20c*	SR	Ch Cal-R		3.3.1	SR 3.3.1.11		
F-Unit 20c*	SR	COT-R		3.3.1	SR 3.3.1.13		
Reactor Trip System Interlock- Power Range Neutron Flux, P-9							
F-Unit 22d	APPLIC.		01-37-A	TBL3.3.1-1	APPLIC.	FUNC. 18d	

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F-Unit 22d	ACTION	8.1(new)	01-12-M	3.3.1	ACTION T		3.3-44
F-Unit 22d	ACTION	8.1(new)	01-05-A	3.3.1	ACTION T		3.3-44
F-Unit 20d*	SR	Ch Cal-R		3.3.1	SR 3.3.1.11		
F-Unit 20d*	SR	COT-R		3.3.1	SR 3.3.1.13		
Reactor Trip System Interlock- Power Range Neutron Flux, P-10							
F-Unit 22e	APPLIC.		01-37-A	TBL3.3.1-1	APPLIC.	FUNC. 18e	
F-Unit 22e	ACTION	8	01-05-A	3.3.1	ACTION S		3.3-44
F-Unit 20e*	SR	Ch Cal-R		3.3.1	SR 3.3.1.11		
F-Unit 20e*	SR	COT-R		3.3.1	SR 3.3.1.13		
Reactor Trip System Interlock- Turbine Impulse Chamber Pressure, P-13 (Input to P-7)							
F-Unit 22f	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 18f	
F-Unit 22f	ACTION	8.1(new)	01-12-M	3.3.1	ACTION T		3.3-44
F-Unit 22f	ACTION	8.1(new)	01-05-A	3.3.1	ACTION T		3.3-44
F-Unit 20f*	SR	Ch Cal-R	01-23-A	3.3.1	SR 3.3.1.11		
F-Unit 20f*	SR	COT-R		3.3.1	SR 3.3.1.13		
F-Unit 23	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 22	3.3-45
F-Unit 23	ACTION	13		3.3.1	ACTION W		
F-Unit 23	SR	Ch Cal-R		3.3.1	SR 3.3.1.12		
F-Unit 23	SR	TADOT-R		3.3.1	SR 3.3.1.14		
F-Unit 23	SR	ALT-R		3.3.1	SR 3.3.1.17		3.3-45
Reactor Trip Bypass Breaker (from Table 4.3-1)							
F-Unit 24	APPLIC.			TBL3.3.1-1	APPLIC.	FUNC. 19	
F-Unit 24	SR	TADOT-M		3.3.1	SR 3.3.1.4		
F-Unit 24	SR	TADOT-R	01-32-LG	3.3.1	SR 3.3.1.4		
Table 3.3-1 notations				3.3.1 various NOTES			
TBL3.3-1	NOTE	*	01-55-LS39	TBL3.3.1-1	NOTE	(b)	3.3-122
TBL3.3-1	NOTE	#	01-05-A				

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TBL3.3-1	NOTE	##		TBL3.3.1-1	NOTE	(c)	
TBL3.3-1	NOTE	###		TBL3.3.1-1	NOTE	(e)	
TBL3.3-1	NOTE	***(new)	01-53-A	TBL3.3.1-1	ACTION D	NOTE	3.3-120
TBL3.3-1	NOTE	(f) New	01-47-A	TBL3.3.1-1	NOTE	(f)	3.3-11
TBL3.3-1	NOTE	(j) New	01-48-LS4	TBL3.3.1-1	NOTE	(j)	
TBL3.3-1	NOTE	(k) New	01-14-A	TBL3.3.1-1	NOTE	(k)	3.3-122
TBL3.3-1	NOTE	(d) New	01-07-LS3	TBL3.3.1-1	NOTE	(d)	
TBL3.3-1	NOTE	(g) New	01-19-LS8	TBL3.3.1-1	NOTE	(g)	
TBL3.3-1	ACTION	1	01-04-LG	3.3.1	ACTION B		3.3-106
Not Needed				3.3.1	ACTION C	NOTE	3.3-135
TBL3.3-1	ACTION	2	01-43-A	3.3.1	ACTION D		
TBL3.3-1	ACTION	2.a		3.3.1	ACTION D	D.1.1	
TBL3.3-1	ACTION	2.b	01-04-LG	3.3.1	ACTION D		
TBL3.3-1	ACTION	2.b	01-17-A	3.3.1	ACTION D	NOTE	
TBL3.3-1	ACTION	2.c	01-18-LS7	3.3.1	ACTION D	D.2.1&2.2	
TBL3.3-1	ACTION	2.c	01-53-A	3.3.1	ACTION D	D.2 NOTE	3.3-120
TBL3.3-1	ACTION	2.c	01-56-A	NA			
TBL3.3-1	ACTION	2(new)	01-18-LS7	3.3.1	ACTION D	D.3	
TBL3.3-1	ACTION	2.1(new)	01-06-LS2	3.3.1	ACTION E		
TBL3.3-1	ACTION	2.1(new)	01-17-A	3.3.1	ACTION E	NOTE	3.3-40
TBL3.3-1	ACTION	3	01-04-LG	3.3.1	ACTION F		
TBL3.3-1	ACTION	3.a	01-07-LS3	3.3.1	APPLIC.	TBL3.3.1-1	
TBL3.3-1	ACTION	3.b	01-07-LS3	3.3.1	ACTION F		3.3-95 3.3-107
TBL3.3-1	ACTION	3.1(new)	01-07-LS3	3.3.1	ACTION G		3.3-95
TBL3.3-1	ACTION	4	01-04-LG	3.3.1	ACTION I		
TBL3.3-1	ACTION	4	01-08-M	3.3.1	ACTION I		

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TBL3.3-1	ACTION	4.1(new)	01-08-M	3.3.1	ACTION J		
TBL3.3-1	ACTION	5	01-04-LG	3.3.1	ACTION L		
TBL3.3-1	ACTION	5	01-09-M	3.3.1	ACTION L		3.3-123
TBL3.3-1	ACTION	6	01-43-A	3.3.1	ACTION M		
TBL3.3-1	ACTION	6.a	01-19-LS8	3.3.1	ACTION M	M.1	
TBL3.3-1	ACTION	6.b	01-04-LG	3.3.1	ACTION M	M.2 & M NOTE	
TBL3.3-1	ACTION	7	01-43-A	3.3.1	ACTION P		
TBL3.3-1	ACTION	7 Note(new)	01-48-LS4	3.3.1	ACTION P	NOTE	3.3-02
TBL3.3-1	ACTION	8	01-04-LG	3.3.1	ACTION S		3.3-44
TBL3.3-1	ACTION	8	01-52-LG	3.3.1	ACTION S		
TBL3.3-1	ACTION	8	01-12-M	3.3.1	ACTION S		
TBL3.3-1	ACTION	8.1(new)	01-12-M	3.3.1	ACTION T		3.3-44
TBL3.3-1	ACTION	9	01-49-LS18	3.3.1	ACTION M		
TBL3.3-1	ACTION	10	01-04-LG	3.3.1	ACTION R	R	
TBL3.3-1	ACTION	10	01-13-LS6	3.3.1	ACTION R	NOTE 1	3.3-43
TBL3.3-1	ACTION	11	01-04-LG	3.3.1	ACTION C		
TBL3.3-1	ACTION	11	01-55-LS39	3.3.1	ACTION C		3.3-122
TBL3.3-1	ACTION	12	01-14-A	3.3.1	ACTION U		3.3-106
TBL3.3-1	ACTION	13	01-43-A	3.3.1	ACTION W		3.3-45
TBL3.3-1	ACTION	13.a	01-04-LG	BASES			
TBL3.3-1	ACTION	13.b		3.3.1	ACTION W	NOTE	
TBL3.3-1	ACTION	26	01-04-LG	3.3.1	ACTION R		
TBL3.3-1	ACTION	27	01-43-A	3.3.1	ACTION X		3.3-46
TBL3.3-1	ACTION	27.a		3.3.1	ACTION X	X.1	
TBL3.3-1	ACTION	27.b	01-43-A	3.3.1	ACTION X	X.2	
TBL3.3-1	ACTION	28	01-50-A	3.3.1	ACTION M		

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Table 4.3-1 notations				3.3.1 various SR NOTES and Table 3.3-1 notes.			
TBL 4.3-1	* note		01-55-LS39	TBL3.3.1-1	note (b)		3.3-122
TBL 4.3-1	## note			TBL3.3.1-1	note (e)		
TBL 4.3-1	### note			TBL3.3.1-1	note (c)		
TBL 4.3-1	note (1)		01-24-LS9	3.3.1	SR 3.3.1.8	NOTE	
TBL 4.3-1	note (1a)		01-24-LS9	3.3.1	SR 3.3.15	NOTE	
TBL 4.3-1	note (2)		01-25-A	3.3.1	SR 3.3.1.2	NOTES1&2	3.3-47
TBL 4.3-1	note (3)		01-25-A	3.3.1	SR 3.3.1.3	NOTE1&2& FREQ.	3.3-96
TBL 4.3-1	note (4)			3.3.1	SR 3.3.1.11	NOTE	
TBL 4.3-1	note (4)			3.3.1	SR 3.3.1.16	NOTE	
TBL 4.3-1	note (5)		01-26-LG	3.3.1	SR 3.3.1.11	NOTE	
TBL 4.3-1	note (6)		01-25-A	3.3.1	SR 3.3.1.6	NOTE	
TBL 4.3-1	note (7)			3.3.1	SR 3.3.1.4	FREQ.	3.3-124
TBL 4.3-1	note (8)		01-28-A	3.3.1	SR 3.3.1.7 & 8	NOTE	3.3-111
TBL 4.3-1	note (9)			3.3.1	SR 3.3.1.9, 14, & 15	NOTE	
TBL 4.3-1	note (10)		01-32-LG	3.3.1	SR 3.3.1.4		
TBL 4.3-1	note (14)		01-32-LG	3.3.1	SR 3.3.1.4		
TBL 4.3-1	note (15)			3.3.1	SR 3.3.1.4	NOTE	
TBL 4.3-1	note (16)		01-32-LG	3.3.1	SR 3.3.1.4		
TBL 4.3-1	note (19)	(new)	01-22-M	3.3.1	SR 3.3.1.8	FREQ.	
TBL 4.3-1	note (19)	(new)	01-27-LS10	3.3.1	SR 3.3.1.8		
TBL 4.3-1	note (20)	(new)	01-22-M	3.3.1	SR 3.3.1.8	NOTE	
TBL 4.3-1	note (22)	(new)	01-23-A	3.3.1	SR 3.3.1.10 & 11		
3.3.2	LCO		02-01-A	3.3.1	LCO		

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3.3.2	LCO		01-35-LG	3.3.2	APPLIC.	TBL3.3.2-1	
3.3.2	APPLIC.	TBL 3.3-3			APPLIC.	TBL3.3.2-1	
3.3.2	ACTION	*Note	01-01-A	3.3.3	ACTION	NOTE	
3.3.2	ACTION	a	02-04-LG	3.3.2	ACTIONS	A thru N	
3.3.2	ACTION	b	02-04-LG	3.3.2	ACTIONS	A thru N	
3.3.2	SR	4.3.2.1		3.3.2	SRS	various	
3.3.2	SR	4.3.2.2	01-02-LG	3.3.2	SR 3.3.2.10		
3.3.2	SR	4.3.2.2	01-03-LS1	3.3.2	ACTIONS		
3.3.2	SR	4.3.2.2	02-40-A	3.3.2	SR 3.3.2.10	NOTE	
3.3.2	New	*Note	01-01-A	3.3.2	ACTIONS	NOTE	
3.3.2	New	**Note	02-40-A	3.3.2	SR 3.3.2.10	NOTE	
TABLE 3.3-3, 3.3-4, and 4.3-2			01-04-LG	TABLE 3.3.2-1, ACTIONS and SRS.			
TABLE 3.3-3, 3.3-4, and 4.3-2			01-43-A	TABLE 3.3.2-1, ACTIONS and SRS.			
TABLE 3.3-3, 3.3-4, and 4.3-2			01-44-A	TABLE 3.3.2-1, ACTIONS and SRS.			
F-Unit 1	APPLIC.		02-19-LG	BASES			
F-Unit 1a	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC. 1a	
F-Unit 1a	ACTION	19		3.3.2	ACTION B		
F-Unit 1a	SR	TADOT-R		3.3.2	SR 3.3.2.8		
F-Unit 1b	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC. 1b	
F-Unit 1b	ACTION	14		3.3.2	ACTION C		
F-Unit 1b	SR	ALT-M		3.3.2	SR 3.3.2.2		
F-Unit 1b	SR	MRT-M		3.3.2	SR 3.3.2.4		
F-Unit 1b	SR	SRT-R		3.3.2	SR 3.3.2.6		
F-Unit 1c	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC. 1c	3.3-66
F-Unit 1c	ACTION	20.2(new)	02-08-M	3.3.2	ACTION D		
F-Unit 1c	SR	Ch Ck-S		3.3.2	SR 3.3.2.1		
F-Unit 1c	SR	Ch Cal-R	01-23-A	3.3.2	SR 3.3.2.9		

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F-Unit 1c	SR	COT-Q		3.3.2	SR 3.3.2.5		
F-Unit 1d	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC. 1d	
F-Unit 1d	ACTION	20		3.3.2	ACTION D		
F-Unit 1d	SR	Ch Ck-S		3.3.2	SR 3.3.2.1		
F-Unit 1d	SR	Ch Cal-R	01-23-A	3.3.2	SR 3.3.2.9		
F-Unit 1d	SR	COT-Q		3.3.2	SR 3.3.2.5		
F-Unit 1f	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC. 1e	
F-Unit 1f	ACTION	20		3.3.2	ACTION D		
F-Unit 1f	SR	Ch Ck-S		3.3.2	SR 3.3.2.1		
F-Unit 1f	SR	Ch Cal-R	01-23-A	3.3.2	SR 3.3.2.9		
F-Unit 1f	SR	COT-Q		3.3.2	SR 3.3.2.5		
F-Unit 2 "	APPLIC.		02-28-LG	TBL3.3.2-1	APPLIC.	FUNC. 2a	
F-Unit 2a	ACTION	19		3.3.2	ACTION B		
F-Unit 2a	SR	TADOT-R		3.3.2	SR 3.3.2.8		
F-Unit 2b	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC. 2b	
F-Unit 2b	ACTION	14		3.3.2	ACTION B		
F-Unit 2b	SR	ALT-M		3.3.2	SR 3.3.2.2		
F-Unit 2b	SR	MRT-M		3.3.2	SR 3.3.2.4		
F-Unit 2b	SR	SRT-R		3.3.2	SR 3.3.2.6		
F-Unit 2c	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC. 2c	3.3-66
F-Unit 2c	ACTION	17		3.3.2	ACTION		
F-Unit 2c	SR	Ch Ck-S		3.3.2	SR 3.3.2.1		
F-Unit 2c	SR	Ch Cal-R	01-23-A	3.3.2	SR 3.3.2.9		
F-Unit 2c	SR	COT-Q		3.3.2	SR 3.3.2.5		
F-Unit 3a1)	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC.3a(1)	
F-Unit 3a1)	ACTION	19		3.3.2	ACTION C		
F-Unit 3a1)	SR	TADOT-R		3.3.2	SR 3.3.2.8		

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F-Unit 3a2)	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC.3a(2)	
F-Unit 3a2)	ACTION	14		3.3.2	ACTION C		
F-Unit 3a2)	SR	ALT-M		3.3.2	SR 3.3.2.2		
F-Unit 3a2)	SR	MRT-M		3.3.2	SR 3.3.2.4		
F-Unit 3a2)	SR	SRT-R		3.3.2	SR 3.3.2.6		
F-Unit 3b1)	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC.3b(1)	
F-Unit 3b1)	ACTION	19		3.3.2	ACTION B		
F-Unit 3b1)	SR	TADOT-R		3.3.2	SR 3.3.2.8		
F-Unit 3b2)	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC.3b(2)	
F-Unit 3b2)	ACTION	14		3.3.2	ACTION C		
F-Unit 3b2)	SR	ALT-M		3.3.2	SR 3.3.2.2		
F-Unit 3b2)	SR	MRT-M		3.3.2	SR 3.3.2.4		
F-Unit 3b2)	SR	SRT-R		3.3.2	SR 3.3.2.6		
F-Unit 3b3)	APPLIC.			TBL3.3.2-1	APPLIC.	FUNC.3b(3)	3.3-66
F-Unit 3b3)	ACTION	17		3.3.2	ACTION C		
F-Unit 3b3)	SR	Ch Ck-S		3.3.2	SR 3.3.2.1		
F-Unit 3b3)	SR	Ch Cal-R	01-23-A	3.3.2	SR 3.3.2.9		
F-Unit 3b3)	SR	COT-Q		3.3.2	SR 3.3.2.5		
F-Unit 3c1)	APPLIC.		02-20-A	3.3.6	APPLIC.		
F-Unit 3c1)	APPLIC.		03-14-LS29	3.3.6	APPLIC.		
F-Unit 3c1)	ACTION	18		3.3.6	ACTION	B&C	
F-Unit 3c1)	ACTION	37(new)	03-14-LS29	3.3.6	ACTION A		
F-Unit 3c1)	SR	ALT-M		3.3.6	SR 3.3.6.2		
F-Unit 3c1)	SR	MRT-M		3.3.6	SR 3.3.6.3		
F-Unit 3c1)	SR	SRT-R		3.3.6	SR 3.3.6.5		
F-Unit 3c4)	APPLIC.		02-20-A	3.3.6	APPLIC.		
F-Unit 3c4)	APPLIC.		03-14-LS29		APPLIC.		

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F-Unit 3c4)	ACTION	18		3.3.6	ACTION	B&C	
F-Unit 3c4)	ACTION	37(new)	03-14-LS29	3.3.6	ACTION A		
F-Unit 3c4)	SR	Ch Ck-S		3.3.6	SR 3.3.6.1		
F-Unit 3c4)	SR	Ch Cal-R	01-23-A	3.3.6	SR 3.3.6.7		
F-Unit 3c4)	SR	COT-Q	02-35-A	3.3.6	SR 3.3.6.4		
F-Unit 4a	APPLIC.		02-07-LS11	F-Unit 4a	APPLIC.	NOTE(i)	
F-Unit 4a	APPLIC.		02-38-LS35	F-Unit 4a	APPLIC.	TBL3.3.2-1	3.3-58
F-Unit 4a	ACTION	24		3.3.2	ACTION	N	
F-Unit 4a	SR	TADOT-R		3.3.2	SR	3.3.2.8	
F-Unit 4b	APPLIC.		02-07-LS11	F-Unit 4b	APPLIC.	TBL3.3.2-1	
F-Unit 4b	ACTION	22		3.3.2	ACTION	G	
F-Unit 4b	SR	ALT-M		3.3.2	SR	3.3.2.2	
F-Unit 4b	SR	MRT-M		3.3.2	SR	3.3.2.4	
F-Unit 4b	SR	SRT-R		3.3.2	SR	3.3.2.6	
F-Unit 4c	APPLIC.		02-07-LS11	F-Unit 4c	APPLIC.	TBL3.3.2-1	
F-Unit 4c	ACTION	17.1(new)	02-15-M	3.3.2	ACTION	E	3.3-66
F-Unit 4c	SR	Ch Ck-S		3.3.2	SR	3.3.2.1	
F-Unit 4c	SR	Ch Cal-R	01-23-A	3.3.2	SR	3.3.2.9	
F-Unit 4c	SR	COT-Q		3.3.2	SR	3.3.2.5	
F-Unit 4d	APPLIC.		02-07-LS11	F-Unit 4c	APPLIC.	TBL3.3.2-1	
F-Unit 4d	ACTION	20		3.3.2	ACTION	E	
F-Unit 4d	SR	Ch Ck-S		3.3.2	SR	3.3.2.1	
F-Unit 4d	SR	Ch Cal-R	01-23-A	3.3.2	SR	3.3.2.9	
F-Unit 4d	SR	COT-Q		3.3.2	SR	3.3.2.5	
F-Unit 4e	APPLIC.		02-07-LS11	F-Unit 4c	APPLIC.	TBL3.3.2-1	
F-Unit 4e	ACTION	20		3.3.2	ACTION	E	
F-Unit 4e	SR	Ch Ck-S		3.3.2	SR	3.3.2.1	

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F-Unit 4e	SR	Ch Cal-R	01-23-A	3.3.2	SR	3.3.2.9	
F-Unit 4e	SR	COT-Q		3.3.2	SR	3.3.2.5	
F-Unit 5a	APPLIC.		02-07-LS11	F-Unit 5a	APPLIC.	TBL3.3.2-1	
F-Unit 5a	ACTION	25		3.3.2	ACTION	H	
F-Unit 5a	SR	ALT-M		3.3.2	SR	3.3.2.2	
F-Unit 5a	SR	MRT-M		3.3.2	SR	3.3.2.4	
F-Unit 5a	SR	SRT-R		3.3.2	SR	3.3.2.6	
F-Unit 5b	APPLIC.		02-07-LS11	F-Unit 5b	APPLIC.	TBL3.3.2-1	
F-Unit 5b	ACTION	35.2(new)	02-08-M	3.3.2	ACTION	H	
F-Unit 5b	SR	Ch Ck-S		3.3.2	SR	3.3.2.1	
F-Unit 5b	SR	Ch Cal-R	01-23-A	3.3.2	SR	3.3.2.9	
F-Unit 5b	SR	COT-Q		3.3.2	SR	3.3.2.5	
F-Unit 6a	APPLIC.			New F/U6a	APPLIC.	TBL3.3.2-1	
F-Unit 6a	ACTION	24		3.3.2	ACTION	N	3.3-58
F-Unit 6a	SR	TADOT-R		3.3.2	SR	3.3.2.8	
F-Unit 6b	APPLIC.			F-Unit 6b	APPLIC.	TBL3.3.2-1	
F-Unit 6b	ACTION	24		3.3.2	ACTION	N	3.3-58
F-Unit 6b	SR	ALT-M		3.3.2	SR	3.3.2.2	
F-Unit 6b	SR	MRT-M		3.3.2	SR	3.3.2.4	
F-Unit 6b	SR	SRT-R		3.3.2	SR	3.3.2.6	
F-Unit 6c1)a	APPLIC.		02-09-LG	F-Unit 6d1)	APPLIC.		
F-Unit 6c1)a	ACTION	20		3.3.2	ACTION	D	
F-Unit 6c1)a	SR	Ch Ck-S		3.3.2	SR	3.3.2.1	
F-Unit 6c1)a	SR	Ch Cal-R	01-23-A	3.3.2	SR	3.3.2.9	
F-Unit 6c1)a	SR	COT-Q		3.3.2	SR	3.3.2.5	
F-Unit 6c1)b	APPLIC.			F-Unit 6d1)	APPLIC.		
F-Unit 6c1)b	ACTION	29		F-Unit 6d1)	ACTION	M	3.3-46

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F-Unit 6c1)b	SR	Ch Cal-R	01-23-A	3.3.2	SR	3.3.2.9	
F-Unit 6c1)b	SR	COT-Q		3.3.2	SR	3.3.2.5	
F-Unit 6c2)	ALL		02-09-LG	Not Used			
F-Unit 6d	APPLIC.		02-19-LG	F-Unit 6g	APPLIC.	TBL3.3.2-1	
F-Unit 6d	ACTION	35	02-19-LG	3.3.2	ACTION	I	3.3-127
F-Unit 6d	SR	Ch Cal-R	01-23-A	3.3.2	SR	3.3.2.9	
F-Unit 6d	SR	TADOT-R		3.3.2	SR	3.3.2.8	
F-Unit 6e	APPLIC.		02-19-LG	BASES	3.3.2		
F-Unit 7a1)	APPLIC.		02-11-A	3.3.5	APPLIC.		
F-Unit 7a1)	APPLIC.		02-36-M	3.8.2	APPLIC.		
F-Unit 7a1)	ACTION	16		3.3.5	ACTION	A	
F-Unit 7a1)	SR	Ch Cal-R	01-23-A	3.3.5	SR	3.3.5.3	
F-Unit 7a1)	SR	TADOT-R		3.3.5	SR	3.3.5.2	
F-Unit 7a2)	APPLIC.		02-11-A	3.3.5	APPLIC.		
F-Unit 7a2)	ACTION	15	02-48-LS28	3.3.5	ACTION	A	
F-Unit 7a2)	SR	Ch Cal-R	01-23-A	3.3.5	SR	3.3.5.3	
F-Unit 7a2)	SR	TADOT-R		3.3.5	SR	3.3.5.2	
F-Unit 7b1)	APPLIC.		02-11-A	3.3.5	APPLIC.		
F-Unit 7b1)	APPLIC.		02-36-M	3.8.2	APPLIC.		
F-Unit 7b1)	ACTION	15	02-48-LS28	3.3.5	ACTION	A	
F-Unit 7b1)	SR	Ch Cal-R	01-23-A	3.3.5	SR	3.3.5.3	
F-Unit 7b1)	SR	TADOT-R		3.3.5	SR	3.3.5.2	
F-Unit 7b2)	APPLIC.		02-11-A	3.3.5	APPLIC.		
F-Unit 7b2)	ACTION	16		3.3.5	ACTION	A	
F-Unit 7b2)	SR	Ch Cal-R	01-23-A	3.3.5	SR	3.3.5.3	
F-Unit 7b2)	SR	TADOT-R		3.3.5	SR	3.3.5.2	
F-Unit 7b3)	APPLIC.		02-11-A	3.3.5	APPLIC.		

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F-Unit 7b3)	ACTION	16		3.3.5	ACTION	A	
F-Unit 7b3)	SR	Ch Cal-R	01-23-A	3.3.5	SR	3.3.5.3	
F-Unit 7b3)	SR	TADOT-R "		3.3.5	SR	3.3.5.2	
F-Unit 8a	APPLIC.		01-37-A	F-Unit 8b	APPLIC.	TBL3.3.2-1	
F-Unit 8a	ACTION	21		3.3.2	ACTION	L	3.3-44
F-Unit 8a	SR	Ch Cal-R	01-23-A	3.3.2	SR	3.3.2.9	
F-Unit 8a	SR	COT-Q		3.3.2	SR	3.3.2.5	
F-Unit 8c	APPLIC.			F-Unit 8a	APPLIC.	TBL3.3.2-1	
F-Unit 8c	ACTION	23		3.3.2	ACTION	F	
F-Unit 8c	SR	TADOT-R			SR	3.3.5.11	
F-Unit9(new)	APPLIC.		02-29-M	F-Unit 7	APPLIC.	TBL3.3.2-1	3.3-29
F-Unit9(new)	ACTION	20.1	02-29-M	3.3.2	ACTION	K	3.3-29
F-Unit9(new)	SR	Ch Ck-S	02-29-M	3.3.2	SR	3.3.2.1	
F-Unit9(new)	SR	Ch Cal-R	02-29-M	3.3.2	SR	3.3.2.9	
F-Unit9(new)	SR	Ch Cal-R	01-23-A	3.3.2	SR	3.3.2.9	
F-Unit9(new)	SR	TADOT-R	02-29-M	3.3.2	SR	3.3.2.8	
TBL3.3-3	NOTE	#		TBL3.3.2-1	NOTE	(b)	3.3-63
TBL3.3-3	NOTE	##		TBL3.3.2-1	NOTE	(g)	3.3-36
TBL3.3-3	NOTE	###		TBL3.3.2-1	NOTE	(k) New	3.3-46
TBL3.3-3	NOTE	** (New)	02-36-M	3.3.5	APPLIC.		
TBL3.3-3	NOTE	(a) (New)	02-07-LS11	TBL3.3.2-1	NOTE	(i)	
TBL3.3-3	NOTE	(b) (New)	02-07-LS11	TBL3.3.2-1	NOTE	(j)	
TBL3.3-3	ACTION	14	01-04-LG	3.3.2	ACTION	C	
TBL3.3-3	ACTION	15	02-48-LS28	3.3.5	ACTION	A	3.3-104
TBL3.3-3	ACTION	15	01-43-A	3.3.5	ACTION	A	3.3-104
TBL3.3-3	ACTION	16	01-43-A	3.3.5	ACTION	A	3.3-104
TBL3.3-3	ACTION	16	02-11-A	3.3.5	ACTION	A	3.3-104

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TBL3.3-3	ACTION	17	01-43-A	3.3.2	ACTION	E	
TBL3.3-3	ACTION	17	02-15-M	3.3.2	ACTION	E	3.3-66
TBL3.3-3	ACTION	17	01-04-LG	3.3.2	ACTION	E	
TBL3.3-3	ACTION	17.1(new)	02-15-M	3.3.2	ACTION	E	3.3-66
TBL3.3-3	ACTION	18	01-04-LG	3.3.6	ACTION	B&C	
TBL3.3-3	ACTION	18	02-05-M	3.3.6	ACTION	B&C	
TBL3.3-3	ACTION	18	02-39-LG	3.3.6	ACTION	B&C	
TBL3.3-3	ACTION	18	03-14-LS29	3.3.6	ACTION	A	
TBL3.3-3	ACTION	19	01-04-LG	3.3.2	ACTION	B	
TBL3.3-3	ACTION	20	02-08-M	3.3.2	ACTION	D	
TBL3.3-3	ACTION	20	01-43-A	3.3.2	ACTION	D	
TBL3.3-3	ACTION	20a		3.3.2	ACTION	D.1	
TBL3.3-3	ACTION	20b	01-04-LG	3.3.2	ACTION	D.2&NOTE	3.3-37
TBL3.3-3	ACTION	20.1(New)	02-29-M	3.3.2	ACTION	K	3.3-29
TBL3.3-3	ACTION	20.2(new)	02-08-M	3.3.2	ACTION	D	3.3-66
TBL3.3-3	ACTION	21	01-04-LG	3.3.2	ACTION	L	
TBL3.3-3	ACTION	21	02-14-M	3.3.2	ACTION	L	3.3-44
TBL3.3-3	ACTION	21	01-52-LG	3.3.2	ACTION	L	
TBL3.3-3	ACTION	22	01-04-LG	3.3.2	ACTION	G	
TBL3.3-3	ACTION	23	01-43-A	3.3.2	ACTION	F	
TBL3.3-3	ACTION	24	01-43-A	3.3.2	ACTION	N	3.3-58
TBL3.3-3	ACTION	25	01-04-LG	3.3.2	ACTION	H	
TBL3.3-3	ACTION	29	01-43-A	3.3.2	ACTION	M	3.3-46
TBL3.3-3	ACTION	29a		3.3.2	ACTION	M	3.3-46
TBL3.3-3	ACTION	29b	01-43-A	3.3.2	ACTION	M	3.3-46
TBL3.3-3	ACTION	35	01-43-A	3.3.2	ACTION	I	
TBL3.3-3	ACTION	35a	02-08-M	3.3.2	ACTION	I	3.3-127

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TBL3.3-3	ACTION	35b	01-04-LG	3.3.2	ACTION	I	3.3-127
TBL3.3-3	ACTION	35.2(New)	02-08-M	3.3.2	ACTION	I	3.3-127
TBL3.3-4	NOTE	1		TBL3.3.2-1	Note	(c)	
TBL3.3-4	NOTE	2		TBL3.3.2-1	Note	(l)	
TBL3.3-4	NOTE	3		TBL3.3.2-1	Note	(h)	
TBL3.3-5			01-35-LG	FSAR			
TBL3.3-5	NOTES		01-35-LG	FSAR			
Table 4.3-2 notes							
TBL 4.3-2	Note	(1)		3.3.2	SRS	3.3.2.2 & 3.3.2.4	
TBL 4.3-2	Note	(2)	02-35-A				
TBL 4.3-2	Note	(3)		TBL3.3.2-1	Note	(g)	3.3-63
TBL 4.3-2	Note	(5)		TBL3.3.2-1	Note	(k)	3.3-46
TBL 4.3-2	Note	(6) New	01-23-A	3.3.2	SR	3.3.2.9	
3.3.3.1	LCO		03-01-A	3.3.6,7&8	LCO		
3.3.3.1	APPLIC.	TBL3.3-6		3.3.6,7&8	APPLIC.		
3.3.3.1	ACTION	a	03-02-M	3.3.6,7&8	ACTIONS	Various	
3.3.3.1	ACTION	b		3.3.6,7&8	ACTIONS	Various	
3.3.3.1	ACTION	c	03-06-A	3.3.8	ACTION	NOTE 2	3.3-34
3.3.3.1	ACTION	* note (new)	01-01-A	3.3.6,7&8	ACTION	NOTE	
3.3.3.1	SR	4.3.3.1		3.3.6,7&8	SRS	Various	
Table 3.3-6 and 4.3-3			01-43-A	LCOs 3.4.15, 3.3.6, 3.3.7, and 3.3.8.			
F-Unit 1	APPLIC.		03-01-A	3.3.6,7,&8	APPLIC.		
F-Ut (new)	APPLIC.		03-08-M	3.3.8	TBL3.3.8-1		
F-Ut (new)	ACTION	32	03-08-M	3.3.8	ACTION	A	3.3-82
F-Ut (new)	SR	TADOT	03-08-M	3.3.8	SR	3.3.8.4	

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F-Ut 1a1)&2)	APPLIC.		03-03-LG	3.3.8	APPLIC.	TBL3.3.8-1	3.3-82
F-Ut 1a1)&2)	APPLIC.		03-07-LS16	3.3.8	APPLIC.	TBL3.3.8-1	3.3-82
F-Ut 1a1)&2)	ACTION	30	01-43-A	3.3.8	ACTION	A	3.3-82
F-Ut 1a1)&2)	ACTION	32	01-43-A	3.3.8	ACTION	A.1.2.3.1	3.3-82
New	SR	TADOT	03-08-M	3.3.8	SR	3.3.8.4	
F-Ut 1a1)&2)	SR	Ch Ck-S		3.3.8	SR	3.3.8.1	
F-Ut 1a1)&2)	SR	Ch Cal-R		3.3.8	SR	3.3.8.5	
F-Ut 1a1)&2)	SR	CFT-Q		3.3.8	SR	3.3.8.2	3.3-75
F-Unit 1b	APPLIC.		03-07-LS16	3.3.8	APPLIC.		
F-Unit 1b	ACTION	32	01-43-A	3.3.8	ACTION	A.1.2.3.1	3.3-82
F-Unit 1b	SR	Ch Ck-S		3.3.8	SR	3.3.8.4	
F-Unit 1b	SR	Ch Cal-R		3.3.8	SR	3.3.8.1	
F-Unit 1b	SR	CFT-Q		3.3.8	SR	3.3.8.5	
F-Unit 2	APPLIC.		03-01-A	3.3.7	APPLIC.		
F-Unit 2	APPLIC.		03-04-M	3.3.7	APPLIC.		
F-Unit 2	APPLIC.			3.3.7	APPLIC.		
F-Unit 2a&b	APPLIC.		03-08-M	3.3.7	APPLIC.		
F-Unit 2a&b	ACTION	34	01-43-A	3.3.7	ACTION	A&C	
F-Unit 2a&b	ACTION	34	03-05-LS14	3.3.7	ACTION	A&C	
F-Unit 2a&b	ACTION	34	03-15-M	3.3.7	ACTION	A&C	
F-Unit 2a&b	ACTION	36	03-15-M	3.3.7	ACTION	B&C	
F-Unit 2a	SR	TADOT	03-08-M	3.3.7	SR	3.3.7.6	
F-Unit 2b	SR	ALT-M	03-08-M	3.3.7	SR	3.3.7.3	
F-Unit 2b	SR	MRT-M	03-08-M	3.3.7	SR	3.3.7.4	
F-Unit 2b	SR	SRT-R	03-08-M	3.3.7	SR	3.3.7.5	
F-Unit 2c	SR	Ch Ck-S	03-08-M	3.3.7	SR	3.3.7.1	
F-Unit 2c	SR	Ch Cal-R	03-08-M	3.3.7	SR	3.3.7.7	

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F-Unit 2c	SR	CFT-Q	03-08-M	3.3.7	SR	3.3.7.2	
F-Unit 3	APPLIC.		03-01-A	3.3.6,3.4.15	APPLIC.		
F-UI(3a2)&b2)	APPLIC.		03-01-A	3.4.15	APPLIC.		
F-UI(3a2)&b2)	ACTION	31	03-0-1-A	3.4.15	ACTION	B&C.1.1	
F-UI(3a2)&b2)	SR	Ch Ck-S	03-0-1-A	3.4.15	SR	3.4.15.1	
F-UI(3a2)&b2)	SR	Ch Cal-R	03-0-1-A	3.4.15	SR	3.4.15.4	
F-UI(3a2)&b2)	SR	CFT-Q	03-0-1-A	3.4.15	SR	3.4.15.2	
F-UI(3a3)&b1)	APPLIC.			3.3.6	APPLIC.		
F-UI(3a3)&b1)	ACTION	33		3.3.6	ACTION	B&C	
F-UI(3a3)&b1)	ACTION	37	03-14-LS29	3.3.6	ACTION	A	
F-UI(3a3)&b1)	SR	Ch Ck-S		3.3.6	SR	3.3.6.1	
F-UI(3a3)&b1)	SR	Ch Cal-R		3.3.6	SR	3.3.6.7	
F-UI(3a3)&b1)	SR	CFT-Q		3.3.6	SR	3.3.6.4	
TBL 3.3-6	NOTE	*	03-03-LG	NA			
TBL 3.3-6	NOTE	**	03-07-LS16	3.3.8	TBL3.3.8-1	NOTE (a)	
TBL 3.3-6	NOTE	***	03-10-LG	NA			
TBL 3.3-6	NOTE	(a)	03-01-A	3.3.8	TBL3.3.8-1	NOTE (b)	
TBL 3.3-6	NOTE	(b)	03-01-A	3.3.8	TBL3.3.8-1	NOTE (c)	
TBL 3.3-6	ACTION	30	01-43-A	3.3.8	ACTION	A&C	3.3-82
TBL 3.3-6	ACTION	31	03-01-A	3.4.15	ACTION	B&C.1.1	
TBL 3.3-6	ACTION	31	01-43-A	3.4.15	ACTION	B&C.1.1	
TBL 3.3-6	ACTION	32	01-43-A	3.3.8	ACTION	A.1.2.3.2	3.3-82
TBL 3.3-6	ACTION	33	01-43-A	3.3.6	ACTION	B&C	
TBL 3.3-6	ACTION	34	01-43-A	3.3.7	ACTION	A&C	
TBL 3.3-6	ACTION	34	03-05-LS14	3.3.7	ACTION	A&C	
TBL 3.3-6	ACTION	34	03-15-M	3.3.7	ACTION	A&C	
TBL 3.3-6	ACTION	36	03-15-M	3.3.7	ACTION	B&C	

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<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
TBL 3.3-6	ACTION	37	03-14-LS29	3.3.6	ACTION	A	
3.3.3.2	LCO		04-01-R				
3.3.3.3	LCO		05-01-R				
3.3.3.4	LCO		06-01-R				
3.3.3.5	LCO			3.3.4	LCO		3.3-94
3.3.3.5	APPLIC.			3.3.4	APPLIC.		
3.3.3.5	ACTION	a.	07-05-A	3.3.4	ACTION	A&B	
3.3.3.5	ACTION	b.		3.3.4	ACTION	NOTE 1	
3.3.3.5	ACTION	c.		3.3.4	ACTION	NOTE 2	
3.3.3.5	SR	4.3.3.5.1		3.3.4	SR	3.3.4.1& 3	
3.3.3.5	SR	4.3.3.5.2		3.3.4	SR	3.3.4.2	
3.3.3.5	TBL 3.3-9		07-06-LG	3.3.4	TBL3.3.4-1		
3.3.3.5	TBL 3.3-9	Inst/CF 1		3.3.4	TBL3.3.4-1	F/Inst/C 1b	
3.3.3.5	TBL 3.3-9	Inst/CF 2		3.3.4	TBL3.3.4-1	F/Inst/C 2a	
3.3.3.5	TBL 3.3-9	Inst/CF 3		3.3.4	TBL3.3.4-1	F/Inst/C 4a	
3.3.3.5	TBL 3.3-9	Inst/CF 4		3.3.4	TBL3.3.4-1	F/Inst/C 3d	
3.3.3.5	TBL 3.3-9	Inst/CF 5	07-10-LS26	3.3.4	TBL3.3.4-1	F/Inst/C 3e	
3.3.3.5	TBL 3.3-9	Inst/CF 6		3.3.4	TBL3.3.4-1	F/Inst/C 3f	
3.3.3.5	TBL 3.3-9	Inst/CF 7	07-10-LS26	3.3.4	TBL3.3.4-1	F/Inst/C 3e	
3.3.3.5	TBL 3.3-9	Inst/CF 8		3.3.4	TBL3.3.4-1	F/Inst/C 4c	
3.3.3.5	TBL 3.3-9	Inst/CF 9		3.3.4	TBL3.3.4-1	F/Inst/C 3a&b	
3.3.3.5	TBL 3.3-9	Inst/CF 10		3.3.4	TBL3.3.4-1	F/Inst/C 3c	
3.3.3.5	TBL 3.3-9	Inst/CF 11		3.3.4	TBL3.3.4-1	F/Inst/C 4b	
3.3.3.5	TBL 3.3-9	Inst/CF 12		3.3.4	TBL3.3.4-1	F/Inst/C 3g	
3.3.3.5	TBL 3.3-9	Inst/CF 13		3.3.4	TBL3.3.4-1	F/Inst/C 3h	
3.3.3.5	TBL 3.3-9	Inst/CF 14		3.3.4	TBL3.3.4-1	F/Inst/C 3i	

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<u>Current Technical Specifications</u>				<u>Improved Technical Specifications</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
TBL 4.3-6	SR	Ch Ck-M		3.3.4	SR	3.3.4.1	3.3-69
TBL 4.3-6	SR	Ch Cal-R		3.3.4	SR	3.3.4.3	
3.3.3.6	LCO		08-01-A	3.3.3	LCO		
3.3.3.6	APPLIC.			3.3.3	APPLIC.		
3.3.3.6	ACTION	a	08-11-LS30	3.3.3	ACTION	A&B	3.3-71
3.3.3.6	ACTION	b	08-11-LS30	3.3.3	ACTION	C&E	3.3-71
3.3.3.6	ACTION	c	08-11-LS30	3.3.3	ACTION	F	
3.3.3.6	ACTION	d	08-11-LS30	3.3.3	ACTION	G	
3.3.3.6	ACTION	d	08-04-LS17	3.3.3	ACTION	G	
3.3.3.6	ACTION	e		3.3.3	ACTION	NOTE 1	
3.3.3.6	ACTION	f (new)	01-01-A	3.3.3	ACTION	NOTE 2	
3.3.3.6	SR		08-11-LS30	3.3.3	SR	3.3.3.1&2	
3.3.3.6	TBL3.3-10		08-01-A	3.3.3	TBL3.3.3-1		
3.3.3.6	TBL3.3-10		08-03-A	3.3.3	TBL3.3.3-1		
3.3.3.6	TBL3.3-10		08-11-LS30	3.3.3	TBL3.3.3-1		
3.3.3.6	TBL3.3-10	Funct. 1		3.3.3	TBL3.3.3-1	Funct 8b)	3.3-71
3.3.3.6	TBL3.3-10	(New)	08-11-LS30	3.3.3	TBL3.3.3-1	Funct 8a)	
3.3.3.6	TBL3.3-10	Funct. 2		3.3.3	TBL3.3.3-1	Funct 3	3.3-71
3.3.3.6	TBL3.3-10	Funct. 3		3.3.3	TBL3.3.3-1	Funct 4	3.3-71
3.3.3.6	TBL3.3-10	Funct. 4		3.3.3	TBL3.3.3-1	Funct 5	
3.3.3.6	TBL3.3-10	Funct. 5		3.3.3	TBL3.3.3-1	Funct 12	
3.3.3.6	TBL3.3-10	Funct. 6		3.3.3	TBL3.3.3-1	Funct 2	3.3-71
3.3.3.6	TBL3.3-10	Funct. 7		3.3.3	TBL3.3.3-1	Funct 13b)	3.3-71
3.3.3.6	TBL3.3-10	(New)	08-11-LS30	3.3.3	TBL3.3.3-1	Funct 13a)	3.3-71
3.3.3.6	TBL3.3-10	Funct. 8		3.3.3	TBL3.3.3-1	Funct (new)	3.3-71
3.3.3.6	TBL3.3-10	Funct. 9		3.3.3	TBL3.3.3-1	Funct 7b)	3.3-71

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<u>Current Technical Specifications</u>				<u>Improved Technical Specifications</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.3.3.6	TBL3.3-10	Funct. 10	08-11-LS30	3.3.3	TBL3.3.3-1	Funct 7a)	3.3-71
3.3.3.6	TBL3.3-10	Funct. 11		3.3.3	TBL3.3.3-1	Funct 19)	3.3-71
3.3.3.6	TBL3.3-10	Funct. 12	08-11-LS30	NA			
3.3.3.6	TBL3.3-10	Funct. 13	08-11-LS30	NA			
3.3.3.6	TBL3.3-10	Funct. 14	08-11-LS30	NA			
3.3.3.6	TBL3.3-10	Funct. 15	08-11-LS30	NA			
3.3.3.6	TBL3.3-10	Funct. 16	08-11-LS30	3.3.3	TBL3.3.3-1	Funct 15,16, 17, & 18	3.3-71
3.3.3.6	TBL3.3-10	Funct. 17	08-11-LS30	NA			
3.3.3.6	TBL3.3-10	Funct. 18		3.3.3	TBL3.3.3-1	Funct 10)	3.3-71
3.3.3.6	TBL3.3-10	Funct. 19	08-11-LS30	NA			
3.3.3.6	TBL3.3-10	Funct. 20		3.3.3	TBL3.3.3-1	Funct 6)	3.3-71
3.3.3.6	TBL3.3-10	(New)	08-11-LS30	3.3.3	TBL3.3.3-1	Funct 1)	3.3-71
3.3.3.6	TBL3.3-10	(New)	08-11-LS30	3.3.3	TBL3.3.3-1	Funct 9)	3.3-71
3.3.3.6	TBL3.3-10	(New)	08-11-LS30	3.3.3	TBL3.3.3-1	Funct 11)	3.3-71
3.3.3.6	TBL3.3-10	(New)	08-11-LS30	3.3.3	TBL3.3.3-1	Funct 14)	3.3-71
3.3.3.6	TBL3.3-10	* NOTE	08-11-LS30	NA			
3.3.3.6	TBL3.3-10	** NOTE	08-11-LS30	NA			
3.3.3.6	TBL3.3-10	*** NOTE	08-11-LS30	NA			
3.3.3.6	TBL3.3-10	NOTE (a)	08-11-LS30	3.3.3	TBL3.3.3-1	NOTE (a)	
3.3.3.6	TBL3.3-10	NOTE (b)	08-11-LS30	3.3.3	TBL3.3.3-1	NOTE (b)	
3.3.3.6	TBL3.3-10	NOTE (c)	08-11-LS30	3.3.3	SR3.3.3.2	NOTE 1	
3.3.3.6	TBL3.3-10	NOTE (d)	08-11-LS30	3.3.3	TBL3.3.3-1	NOTE (c)	3.3-71
3.3.3.6	TBL4.3-7		08-11-LS30	3.3.3	SRS	3.3.3.1&2	
3.3.3.6	TBL4.3-7	SR Ch Ck-M	08-11-LS30	3.3.3	SR	3.3.3.1	
3.3.3.6	TBL4.3-7	SR ChCal-R	08-11-LS30	3.3.3	SR	3.3.3.2	

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<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.3.3.6	TBL4.3-7	NOTE *		3.3.3	SR3.3.3.2	NOTE 2	3.3-20
3.3.3.7	LCO		11-01-R	NA			
3.3.3.10	LCO		09-01-LG	NA			
3.3.4.1	LCO		10-01-R	NA			

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Item	Code	Para.	Change	Item	Code	Para.	Change
3.3.1	LCO		01-35-LG	3.3.1	LCO		
3.3.1	APPLIC.	TBL3.3-1		3.3.1	APPLIC.		
3.3.1	ACTION	New*Note	01-01-A	3.3.1	ACTION	NOTE	
3.3.1	TBL3.3-1	ACTIONS		3.3.1	ACTION	A	
3.3.1	ACTION	1		3.3.1	ACTION	B	3.3-106
3.3.1	TBL3.3-1	Note (e)	01-60-A	3.3.1	ACTION	C NOTE	3.3-135
TBL3.3-1	ACTION	11		3.3.1	ACTION	C	3.3-122
TBL3.3-1	ACTION	2.b	01-17-A	3.3.1	ACTION	D.1.1 NOTE	
TBL3.3-1	ACTION	2.b	01-04-A	3.3.1	ACTION	D	
TBL3.3-1	ACTION	2.c	01-53-A	3.3.1	ACTION	D.2.2 NOTE	3.3-120
TBL3.3-1	ACTION	2.1(new)	01-07-A	3.3.1	ACTION	E.1 NOTE	3.3-40
TBL3.3-1	ACTION	2.1(new)	01-06-LS2	3.3.1	ACTION	E	
TBL3.3-1	ACTION	3b	01-07-LS3	3.3.1	ACTION	F	3.3-95
TBL3.3-1	ACTION	3b	01-07-LS3	3.3.1	ACTION	F	3.3-107
TBL3.3-1	ACTION	3.1(new)	01-07-LS3	3.3.1	ACTION	G	3.3-95
Not Used				3.3.1	ACTION	H	3.3-95
TBL3.3-1	ACTION	4	01-08-M	3.3.1	ACTION	I	
TBL3.3-1	ACTION	4.1	01-08-M	3.3.1	ACTION	J	
TBL3.3-1	ACTION	11	01-55-LS39	3.3.1	ACTION	K	3.3-122
TBL3.3-1	ACTION	5	01-09-M	3.3.1	ACTION	L	3.3-123
TBL3.3-1	ACTION	6b	01-04-LG	3.3.1	ACTION	M NOTE	
TBL3.3-1	ACTION	6	01-19-LS8	3.3.1	ACTION	M	
Not Used				3.3.1	ACTION N	Not Used	3.3-42
Not Used				3.3.1	ACTION O	Not Used	3.3-103
TBL3.3-1	ACTION	7 Note	01-48-LS4	3.3.1	ACTION	P.1 NOTE	3.3-02
TBL3.3-1	ACTION	7	01-43-A	3.3.1	ACTION	P	
TBL3.3-1	ACTION	26		3.3.1	ACTION	Q.1 NOTE	
TBL3.3-1	ACTION	26	01-04-LG	3.3.1	ACTION	Q	

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<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
TBL3.3-1	ACTION	10	01-13-LS6	3.3.1	ACTION	R.1 NOTE1	3.3-43
TBL3.3-1	ACTION	12		3.3.1	ACTION	R.1 NOTE2	3.3-43
TBL3.3-1	ACTION	12		3.3.1	ACTION	R.1 NOTE2	3.3-117
TBL3.3-1	ACTION	26		3.3.1	ACTION	R.1 NOTE3	3.3-117
TBL3.3-1	ACTION	26		3.3.1	ACTION	R.1 NOTE3	3.3-03
TBL3.3-1	ACTION	10	01-13-LS6	3.3.1	ACTION	R	
TBL3.3-1	ACTION	8	01-12-M	3.3.1	ACTION	S	3.3-44
TBL3.3-1	ACTION	8.1	01-12-M	3.3.1	ACTION	T	3.3-44
TBL3.3-1	ACTION	12	01-14-A	3.3.1	ACTION	U	3.3-106
Not Used				3.3.1	ACTION V	Not Used	3.3-93
TBL3.3-1	ACTION	13.b	01-04-LG	3.3.1	ACTION	W.1 NOTE	3.3-45
TBL3.3-1	ACTION	13	01-43-A	3.3.1	ACTION	W	3.3-45
TBL3.3-1	ACTION	27	01-43-A	3.3.1	ACTION	X	3.3-46
TBL4.3-1	SR	Ch Ck-S		3.3.1	SR 3.3.1.1		
TBL4.3-1	Note (2)		01-25-A	3.3.1	SR 3.3.1.2	NOTE 1	
TBL4.3-1	Note (2)		01-25-A	3.3.1	SR 3.3.1.2	NOTE 2	3.3-47
TBL4.3-1	SR	Ch Cal-D	01-21-A	3.3.1	SR 3.3.1.2		
TBL4.3-1	Note (3)		01-25-A	3.3.1	SR 3.3.1.3	NOTE 1	
TBL4.3-1	Note (3)		01-25-A	3.3.1	SR 3.3.1.3	NOTE 2	3.3-96
TBL4.3-1	SR	Ch Cal-M	01-21-A	3.3.1	SR 3.3.1.3		
TBL4.3-1	Note (15)			3.3.1	SR 3.3.1.4	NOTE	3.3-124
TBL4.3-1	SR	TADOT-M	01-32-LG	3.3.1	SR 3.3.1.4		
TBL4.3-1	SR	ALT-M		3.3.1	SR 3.3.1.5		
TBL4.3-1	Note (6)		01-25-A	3.3.1	SR 3.3.1.6	NOTE	3.3-06
TBL4.3-1	SR	Ch Cal-Q		3.3.1	SR 3.3.1.6		
TBL4.3-1	Note (19)		01-27-LS10	3.3.1	SR 3.3.1.7	NOTE 1	
TBL4.3-1	Note (20)		01-22-M	3.3.1	SR 3.3.1.7	NOTE 2	3.3-111
TBL4.3-1	SR	COT-Q		3.3.1	SR 3.3.1.7		

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TBL4.3-1	Note (20)		01-22-M	3.3.1	SR 3.3.1.8	NOTE	
TBL4.3-1	SR	COT-S/U		3.3.1	SR 3.3.1.8		3.3-49
TBL4.3-1	Note (19)		01-27-LS10	3.3.1	SR 3.3.1.8	FREQ.- NOTE	
TBL4.3-1	Note (9)			3.3.1	SR 3.3.1.9	NOTE	
TBL4.3-1	SR	TADOT-Q		3.3.1	SR 3.3.1.9		
TBL4.3-1	Note (2)		01-25-A	3.3.1	SR3.3.1.10	NOTE	
TBL4.3-1	SR	Ch Cal-R	01-23-A	3.3.1	SR3.3.1.10		
TBL4.3-1	Note (4)		01-25-A	3.3.1	SR3.3.1.11	NOTE 1	
TBL4.3-1	Note (22)		01-23-A	3.3.1	SR3.3.1.11	NOTE 2	3.3-125
TBL4.3-1	Note (5)			3.3.1	SR3.3.1.11	NOTE 3	3.3-07
TBL4.3-1	SR	Ch Cal-R	01-23-A	3.3.1	SR3.3.1.11		
Not Used				3.3.1	SR3.3.1.12	NOTE	3.3-101
TBL4.3-1	SR	Ch Cal-R		3.3.1	SR3.3.1.12		
TBL4.3-1	SR	COT-R		3.3.1	SR3.3.1.13		
TBL4.3-1	Note (9)		01-25-A	3.3.1	SR3.3.1.14	NOTE	
TBL4.3-1	SR	TADOT-R		3.3.1	SR3.3.1.14		
TBL4.3-1	Note (9)		01-25-A	3.3.1	SR3.3.1.15	NOTE	
TBL4.3-1	SR	TADOT-S/U		3.3.1	SR3.3.1.15		
TBL4.3-1	Note (1a)		01-24-LS9	3.3.1	SR3.3.1.15	FREQ- NOTE	
TBL3.3-2	Note (1)			3.3.1	SR3.3.1.16	NOTE	
TBL4.3-1	SR	4.3.1.2	01-03-LS1	3.3.1	SR3.3.1.16		
TBL4.3-1	SR	ALT-R		3.3.1	SR3.3.1.17		3.3-45
TBL3.3-1	F-Unit 1			3.3.1	TBL3.3.1-1	FUNC. 1	
TBL3.3-1	F-Unit 2a			3.3.1	TBL3.3-1	FUNC. 2a	
TBL3.3-1	F-Unit 2b			3.3.1	TBL3.3-1	FUNC. 2b	
TBL3.3-1	F-Unit 3			3.3.1	TBL3.3-1	FUNC. 3a	

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<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
TBL3.3-1	F-Unit 4			3.3.1	TBL3.3-1	FUNC. 3b	
TBL3.3-1	F-Unit 5			3.3.1	TBL3.3-1	FUNC. 4	3.3-95
TBL3.3-1	F-Unit 6			3.3.1	TBL3.3-1	FUNC. 5	
TBL3.3-1	F-Unit 7			3.3.1	TBL3.3-1	FUNC. 6	3.3-101
TBL3.3-1	F-Unit 8			3.3.1	TBL3.3-1	FUNC. 7	3.3-101
TBL3.3-1	F-Unit 9			3.3.1	TBL3.3-1	FUNC. 8a	
TBL3.3-1	F-Unit 10			3.3.1	TBL3.3-1	FUNC. 8b	
TBL3.3-1	F-Unit 11			3.3.1	TBL3.3-1	FUNC. 9	
TBL3.3-1	F-Unit 12		01-19-LS8	3.3.1	TBL3.3-1	FUNC10	3.3-09
TBL3.3-1	F-Ut 12a		01-57-LG	3.3.1	TBL3.3-1	FUNC10a	3.3-09
TBL3.3-1	F-Ut 12b		01-57-LG	3.3.1	TBL3.3-1	FUNC10b	3.3-42
TBL3.3-1	F-Unit 19			3.3.1	TBL3.3-1	FUNC11	3.3-103
Not Used				3.3.1	TBL3.3-1	FUNC11a	3.3-103
Not Used				3.3.1	TBL3.3-1	FUNC11b	3.3-103
TBL3.3-1	F-Unit 15			3.3.1	TBL3.3-1	FUNC12	
TBL3.3-1	F-Unit 16			3.3.1	TBL3.3-1	FUNC13	
TBL3.3-1	F-Unit 13			3.3.1	TBL3.3-1	FUNC14	
TBL3.3-1	F-Ut 13a			3.3.1	TBL3.3-1	FUNC14a	3.3-46
TBL3.3-1	F-Ut 13b			3.3.1	TBL3.3-1	FUNC14b	3.3-46
Not Used				3.3.1	TBL3.3-1	FUNC15	3.3-01
TBL3.3-1	F-Ut 17a			3.3.1	TBL3.3-1	FUNC16a	
TBL3.3-1	F-Ut 17b			3.3.1	TBL3.3-1	FUNC16b	
TBL3.3-1	F-Unit 18			3.3.1	TBL3.3-1	FUNC17	
TBL3.3-1	F-Ut 22a			3.3.1	TBL3.3-1	FUNC18a	
TBL3.3-1	F-Ut 22b			3.3.1	TBL3.3-1	FUNC18b	3.3-54
TBL3.3-1	F-Ut 22c			3.3.1	TBL3.3-1	FUNC18c	3.3-44
TBL3.3-1	F-Ut 22d			3.3.1	TBL3.3-1	FUNC18d	3.3-44
TBL3.3-1	F-Ut 22e			3.3.1	TBL3.3-1	FUNC18e	3.3-44

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Current Specs				Improved Specs			
Item	Code	Para.	Change	Item	Code	Para.	Change
TBL3.3-1	F-Ut 22f			3.3.1	TBL3.3-1	FUNC18f	
TBL3.3-1	F-Unit 20			3.3.1	TBL3.3-1	FUNC19	
TBL3.3-1	F-U(new)			3.3.1	TBL3.3-1	FUNC20	3.3-124
TBL3.3-1	F-Unit 21			3.3.1	TBL3.3-1	FUNC 21	
TBL3.3-1	F-Unit 23			3.3.1	TBL3.3-1	FUNC 22	3.3-45
NA				3.3.1	TBL3.3-1	NOTE (a)	Ed
TBL3.3-1	NOTE	*	01-55-LS39	3.3.1	TBL3.3-1	NOTE (b)	3.3-122
TBL3.3-1	NOTE	##		3.3.1	TBL3.3-1	NOTE (c)	
TBL3.3-1	NOTE	(d) New		3.3.1	TBL3.3-1	NOTE (d)	
TBL3.3-1	NOTE	###		3.3.1	TBL3.3-1	NOTE (e)	3.3-95
TBL3.3-1	NOTE	(f) New		3.3.1	TBL3.3-1	NOTE (f)	3.3-11
TBL3.3-1	NOTE	(g) New		3.3.1	TBL3.3-1	NOTE (g)	
Not Used				3.3.1	TBL3.3-1	NOTE (h)	3.3-42
Not Used				3.3.1	TBL3.3-1	NOTE (i)	3.3-42
TBL3.3-1	NOTE	(j) New		3.3.1	TBL3.3-1	NOTE (j)	
TBL3.3-1	NOTE	(k) New	01-14-A	3.3.1	TBL3.3-1	NOTE (k)	
2.2.1	TBL2.2-1	** Note		3.3.1	TBL3.3-1	NOTE (l)	3.3-09
2.2.1	TBL2.2-1	NOTE 1		3.3.2	TBL3.3-1	Note (1)	3.3-10
2.2.1	TBL2.2-1	NOTE 1		3.3.2	TBL3.3-1	Note (1)	3.3-13
2.2.1	TBL2.2-1	NOTE 3		3.3.2	TBL3.3-1	Note (2)	3.3-10
2.2.1	TBL2.2-1	NOTE 3		3.3.2	TBL3.3-1	Note (2)	3.3-13
2.2.1	TBL2.2-1	NOTE 5		3.3.2	TBL3.3-1	Note (3)	3.3-46
3.3.2	LCO		02-01-A	3.3.2	LCO		
3.3.2	APPLIC.			3.3.2	APPLIC.		
3.3.2	ACTION	New*Note		3.3.2	ACTION	NOTE	
3.3.2	TBL3.2-1	ACTIONS		3.3.2	ACTION	A	
3.3.2	ACTION	19	01-04-LG	3.3.2	ACTION	B	

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<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.3.2	ACTION	14	01-04-LG	3.3.2	ACTION	C.1 NOTE	
3.3.2	ACTION	14		3.3.2	ACTION	C	
3.3.2	ACTION	20b	01-04-LG	3.3.2	ACTION	D.1NOTE	3.3-37
3.3.2	ACTION	20	02-08-M	3.3.2	ACTION	D	
3.3.2	ACTION	17		3.3.2	ACTION	E.1 NOTE	
3.3.2	ACTION	17	02-15-M	3.3.2	ACTION	E	3.3-66
3.3.2	ACTION	23	01-43-A	3.3.2	ACTION	F	
3.3.2	ACTION	22		3.3.2	ACTION	G.1 NOTE	
3.3.2	ACTION	22	01-04-LG	3.3.2	ACTION	G	
3.3.2	ACTION	25		3.3.2	ACTION	H.1 NOTE	
3.3.2	ACTION	25	01-04-LG	3.3.2	ACTION	H	
3.3.2	ACTION	35.2b	02-08-M	3.3.2	ACTION	I.1 NOTE	
3.3.2	ACTION	35.2	02-08-M	3.3.2	ACTION	I	3.3-127
Not Used				3.3.2	ACTION	J	3.3-116
3.3.2	ACTION	20.1b	02-29-M	3.3.2	ACTION	K.1.1 NOTE	
3.3.2	ACTION	20.1	02-29-M	3.3.2	ACTION	K	3.3-29
3.3.2	ACTION	21	02-14-M	3.3.2	ACTION	L	3.3-44
3.3.2	ACTION	29	01-43-A	3.3.2	ACTION	M	3.3-46
3.3.2	ACTION	24	01-43-A	3.3.2	ACTION	N	3.3-58
3.3.2	SR	4.3.2.2		3.3.2	SR	NOTE	
TBL 4.3-2	SR	Ch Ck-S		3.3.2	SR 3.3.2.1		
TBL 4.3-2	SR	ALT-M		3.3.2	SR 3.3.2.2		
Not Used				3.3.2	SR 3.3.2.3		3.3-60
TBL 4.3-2	SR	MRT-M		3.3.2	SR 3.3.2.4		
TBL 4.3-2	SR	COT-Q		3.3.2	SR 3.3.2.5		
TBL 4.3-2	SR	SRT-M		3.3.2	SR 3.3.2.6		
Not Used				3.3.2	SR 3.3.2.7		3.3-60
TBL 4.3-2	SR			3.3.2	SR 3.3.2.8	NOTE	

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<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
TBL 4.3-2	SR	TADOT-R		3.3.2	SR 3.3.2.8		
TBL 4.3-2	SR	Note (6)		3.3.2	SR 3.3.2.9	NOTE	
TBL 4.3-2	SR	Ch Cal-R		3.3.2	SR 3.3.2.9		
3.7.1.2	4.7.1.2.1b	* Note		3.3.2	SR3.3.2.10	NOTE	
3.3.2	4.3.2.2	RTT		3.3.2	SR3.3.2.10		
Not Used				3.3.2	SR3.3.2.11	NOTE	
TBL 4.3-2	SR	TADOT-R		3.3.2	SR3.3.2.11		3.3-61
3.3.2	TBL 3.3-3	F-Unit 1a		3.3.2	TBL3.3.2-1	FUNC 1a	
3.3.2	TBL 3.3-3	F-Unit 1b		3.3.2	TBL3.3.2-1	FUNC 1b	
3.3.2	TBL 3.3-3	F-Unit 1c		3.3.2	TBL3.3.2-1	FUNC 1c	3.3-66
Not Used				3.3.2	TBL3.3.2-1	FUNC 1c	3.3-01
3.3.2	TBL 3.3-3	F-Unit 1d		3.3.2	TBL3.3.2-1	FUNC 1d	
3.3.2	TBL 3.3-3	F-Unit 1f		3.3.2	TBL3.3.2-1	FUNC1e1	3.3-66
NA				3.3.2	TBL3.3.2-1	FUNC1e2	3.3-01
NA				3.3.2	TBL3.3.2-1	FUNC 1f	3.3-01
NA				3.3.2	TBL3.3.2-1	FUNC 1g	3.3-01
3.3.2	TBL 3.3-3	F-Unit 2a		3.3.2	TBL3.3.2-1	FUNC 2a	3.3-53
3.3.2	TBL 3.3-3	F-Unit 2b		3.3.2	TBL3.3.2-1	FUNC 2b	
3.3.2	TBL 3.3-3	F-Unit 2c		3.3.2	TBL3.3.2-1	FUNC 2c	3.3-66
NA				3.3.2	TBL3.3.2-1	FUNC 2c	3.3-01
3.3.2	TBL 3.3-3	F-Unit 3a1		3.3.2	TBL3.3.2-1	FUNC3a(1)	
3.3.2	TBL 3.3-3	F-Unit 3a2		3.3.2	TBL3.3.2-1	FUNC3a(2)	
3.3.2	TBL 3.3-3	F-Unit 3a3		3.3.2	TBL3.3.2-1	FUNC3a(3)	
3.3.2	TBL 3.3-3	F-Unit 3b1		3.3.2	TBL3.3.2-1	FUNC3b(1)	3.3-53
3.3.2	TBL 3.3-3	F-Unit 3b2		3.3.2	TBL3.3.2-1	FUNC3b(2)	
3.3.2	TBL 3.3-3	F-Unit 3b3		3.3.2	TBL3.3.2-1	FUNC3b(3)	3.3-66
3.3.2	TBL 3.3-3	F-Unit 4a		3.3.2	TBL3.3.2-1	FUNC4a	3.3-58
3.3.2	TBL 3.3-3	F-Unit 4b		3.3.2	TBL3.3.2-1	FUNC4b	

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<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.3.2	TBL 3.3-3	F-Unit 4c		3.3.2	TBL3.3.2-1	FUNC4c	3.3-137
3.3.2	TBL 3.3-3	F-Unit 4d		3.3.2	TBL3.3.2-1	FUNC4d(1)	
3.3.2	TBL 3.3-3	F-Unit 4e		3.3.2	TBL3.3.2-1	FUNC4d(2)	
NA				3.3.2	TBL3.3.2-1	FUNC4e	3.3-01
NA				3.3.2	TBL3.3.2-1	FUNC4f	3.3-01
NA				3.3.2	TBL3.3.2-1	FUNC4g	3.3-01
NA				3.3.2	TBL3.3.2-1	FUNC4h	3.3-01
3.3.2	TBL 3.3-3	F-Unit 5a		3.3.2	TBL3.3.2-1	FUNC5a	
3.3.2	TBL 3.3-3	F-Unit 5b		3.3.2	TBL3.3.2-1	FUNC5b	
3.3.2	TBL 3.3-3	F-Unit 1		3.3.2	TBL3.3.2-1	FUNC5c	
3.3.2	TBL 3.3-3	F-Unit 6a		3.3.2	TBL3.3.2-1	FUNC6a	3.3-58
3.3.2	TBL 3.3-3	F-Unit 6b		3.3.2	TBL3.3.2-1	FUNC6b	
NA				3.3.2	TBL3.3.2-1	FUNC6c	3.3-01
3.3.2	TBL 3.3-3	F-Unit 6c1		3.3.2	TBL3.3.2-1	FUNC6d1)	3.3-46
3.3.2	TBL 3.3-3	F-Unit 6c2		3.3.2	TBL3.3.2-1	FUNC6d2)	
3.3.2	TBL 3.3-3	F-Unit 6e		3.3.2	TBL3.3.2-1	FUNC6e	
NA				3.3.2	TBL3.3.2-1	FUNC6f	3.3-01
3.3.2	TBL 3.3-3	F-Unit 6d		3.3.2	TBL3.3.2-1	FUNC6g	3.3-127
NA				3.3.2	TBL3.3.2-1	FUNC6h	3.3-116
NA				3.3.2	TBL3.3.2-1	FUNC6i	3.3-01
3.3.2	TBL 3.3-3	F-Unit 9		3.3.2	TBL3.3.2-1	FUNC7	3.3-29
Not Used				3.3.2	TBL3.3.2-1	FUNC7a	3.3-29
Not Used				3.3.2	TBL3.3.2-1	FUNC7b	3.3-29
Not Used				3.3.2	TBL3.3.2-1	FUNC7c	3.3-29
3.3.2	TBL 3.3-3	F-Unit 8c		3.3.2	TBL3.3.2-1	FUNC8a	
3.3.2	TBL 3.3-3	F-Unit 8a		3.3.2	TBL3.3.2-1	FUNC8b	3.3-44
3.3.2	TBL 3.3-3	F-Unit 8a		3.3.2	TBL3.3.2-1	FUNC8b	3.3-15
NA				3.3.2	TBL3.3.2-1	FUNC8c	3.3-01

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<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
Not Used				3.3.2	TBL3.3-1	NOTE (a)	Ed
3.3.2	TBL 3.3-3	# note		3.3.2	TBL3.3-1	NOTE (b)	3.3-63
3.3.2	TBL 3.3-4	NOTE 1		3.3.2	TBL3.3-1	NOTE (c)	3.3-105
NA				3.3.2	TBL3.3-1	NOTE (d)	3.3-01
NA				3.3.2	TBL3.3-1	NOTE (e)	3.3-01
NA				3.3.2	TBL3.3-1	NOTE (f)	3.3-01
3.3.2	TBL 3.3-3	## note		3.3.2	TBL3.3-1	NOTE (g)	3.3-63
3.3.2	TBL 3.3-4	NOTE 3		3.3.2	TBL3.3-1	NOTE (h)	3.3-105
3.3.2	TBL 3.3-3	(a) new	02-07-LS11	3.3.2	TBL3.3-1	NOTE (i)	
3.3.2	TBL 3.3-3	(b) new	02-07-LS11	3.3.2	TBL3.3-1	NOTE (j)	
3.3.2	TBL 3.3-3	### note		3.3.2	TBL3.3-1	NOTE (k)	3.3-46
3.3.2	TBL 3.3-4	NOTE 2		3.3.2	TBL3.3-1	NOTE (l)	3.3-46
3.3.3.6	LCO		08-01-A	3.3.3	LCO		
3.3.3.6	APPLIC.			3.3.3	APPLIC.		
3.3.3.6	ACTION	e		3.3.3	ACTIONS	NOTE 1	
3.3.3.6	ACTION	f (new)	01-01-A	3.3.3	ACTIONS	NOTE 2	
3.3.3.6	ACTION	a	08-11-LS30	3.3.3	ACTION	A	3.3-71
3.3.3.6	ACTION	a	08-11-LS30	3.3.3	ACTION	B	
3.3.3.6	ACTION	b	08-11-LS30	3.3.3	ACTION	C NOTE	
3.3.3.6	ACTION	b	08-11-LS30	3.3.3	ACTION	C	3.3-71
3.6.4.1	APPLIC.			3.3.3	ACTION	D NOTE	3.3-68
3.6.4.1	ACTION	b	08-11-LS30	3.3.3	ACTION	D	
3.3.3.6	ACTION	b	08-11-LS30	3.3.3	ACTION	E	
3.3.3.6	ACTION	c	08-11-LS30	3.3.3	ACTION	F	
3.3.3.6	ACTION	d	08-04-LS17	3.3.3	ACTION	G	
3.6.4.1	ACTION	b		3.3.3	ACTION	H	3.3-68
3.3.3.6	SR	4.3.3.6	08-11-LS30	3.3.3	SR	NOTE	

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<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.3.3.6	SR	4.3.3.6	08-11-LS30	3.3.3	SR3.3.3.1		
3.3.3.6	TBL3.3-10	(d) new	08-11-LS30	3.3.3	SR3.3.3.2	NOTE 1	
3.3.3.6	TBL 4.3-7	* note	08-11-LS30	3.3.3	SR3.3.3.2	NOTE 2	3.3-20
3.3.3.6	SR	4.3.3.6	08-11-LS30	3.3.3	SR3.3.3.2		
3.3.3.6	TBL3.3-10	(new)	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 1	3.3-71
Not Used				3.3.3	TBL3.3.3-1	FUNC 2	3.3-01
3.3.3.6	TBL3.3-10	FUNC. 6	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 2	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 2	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 3	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 3	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 4	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 4	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 5	
3.3.3.6	TBL3.3-10	FUNC. 20	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 6	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 10	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 7a	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 9	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 7b	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 1	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 8a	
3.3.3.6	TBL3.3-10	FUNC. 1	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 8b	3.3-71
3.3.3.6	TBL3.3-10	(new)	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 9	
3.3.3.6	TBL3.3-10	FUNC. 18	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 10	
3.6.4.1	LCO/ACT	a & b		3.3.3	TBL3.3.3-1	FUNC 11	
3.3.3.6	TBL3.3-10	FUNC. 5	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC 12	
3.3.3.6	TBL3.3-10	(new)	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC13a	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 7	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC13b	3.3-71
3.3.3.6	TBL3.3-10	(new)	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC14	
3.3.3.6	TBL3.3-10	FUNC. 8	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC(new)	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 16	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC15	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 16	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC16	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 16	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC17	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 16	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC18	3.3-71
3.3.3.6	TBL3.3-10	FUNC. 11	08-11-LS30	3.3.3	TBL3.3.3-1	FUNC19	3.3-71

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<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.3.3.6	TBL3.3-10	(a) new	08-11-LS30	3.3.3	TBL3.3.3-1	Note (a)	
3.3.3.6	TBL3.3-10	(b) new	08-11-LS30	3.3.3	TBL3.3.3-1	Note (b)	
3.3.3.6	TBL3.3-10	(d) new	08-11-LS30	3.3.3	TBL3.3.3-1	Note (c)	3.3-71
3.3.3.5	LCO			3.3.4	LCO		3.3-94
3.3.3.5	APPLIC.			3.3.4	APPLIC.		
3.3.3.5	ACTION	b		3.3.4	ACTION	NOTE 1	
3.3.3.5	ACTION	c		3.3.4	ACTION	NOTE 2	
3.3.3.5	ACTION	a		3.3.4	ACTION	A	
3.3.3.5	ACTION	a		3.3.4	ACTION	B	
3.3.3.5	TBL4.3-6	Ch Ck		3.3.4	SR 3.3.4.1	NOTE	3.3-22
3.3.3.5	SR	4.3.3.5.1		3.3.4	SR 3.3.4.1		3.3-69
3.3.3.5	SR	4.3.3.5.2		3.3.4	SR 3.3.4.2		
NA				3.3.4	SR 3.3.4.3	NOTE	3.3-84
3.3.3.5	TBL4.3-6	Ch Cal		3.3.4	SR 3.3.4.3	NOTE	3.3-22
3.3.3.5	SR	4.3.3.5.1		3.3.4	SR 3.3.4.3		
3.3.3.5	TBL 3.3-9	07-06-LG		3.3.4	TBL3.3.4-1		3.3-128
Not Used				3.3.4	TBL3.3.4-1	F//C 1.a	3.3-128
3.3.3.5	TBL 3.3-9	INST/CF 1		3.3.4	TBL3.3.4-1	F//C 1.b	
Not Used				3.3.4	TBL3.3.4-1	F//C 1.c	3.3-128
3.3.3.5	TBL 3.3-9	INST/CF 2		3.3.4	TBL3.3.4-1	F//C 2.a	
Not Used				3.3.4	TBL3.3.4-1	F//C 2.b	3.3-128
3.3.3.5	TBL 3.3-9	INST/CF 9		3.3.4	TBL3.3.4-1	F//C 3.a	
3.3.3.5	TBL 3.3-9	INST/CF 9,		3.3.4	TBL3.3.4-1	F//C 3.b	
3.3.3.5	TBL 3.3-9	INST/CF10		3.3.4	TBL3.3.4-1	F//C 3.c	
3.3.3.5	TBL 3.3-9	INST/CF 4		3.3.4	TBL3.3.4-1	F//C 3.d	
3.3.3.5	TBL 3.3-9	INST/CF 5	07-10-LS26	3.3.4	TBL3.3.4-1	F//C 3.e	
3.3.3.5	TBL 3.3-9	INST/CF 6		3.3.4 (New)	TBL3.3.4-1	F//C 3.f	3.3-128

CROSS-REFERENCE TABLE FOR 3/4.3
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.3.3.5	TBL 3.3-9	INST/CF12		3.3.4 (New)	TBL3.3.4-1	F/I/C 3.g	3.3-128
3.3.3.5	TBL 3.3-9	INST/CF13		3.3.4 (New)	TBL3.3.4-1	F/I/C 3.h	3.3-128
3.3.3.5	TBL 3.3-9	INST/CF14		3.3.4 (New)	TBL3.3.4-1	F/I/C 3.i	3.3-128
3.3.3.5	TBL 3.3-9	INST/CF 3		3.3.4	TBL3.3.4-1	F/I/C 4.a	
3.3.3.5	TBL 3.3-9	INST/CF11		3.3.4	TBL3.3.4-1	F/I/C 4.b	
3.3.3.5	TBL 3.3-9	INST/CF 8		3.3.4 (New)	TBL3.3.4-1	F/I/C 4.c	3.3-128
3.3.2	TBL 3.3-3	F-Unit 7	02-11-A	3.3.5	LCO		3.3-133
3.3.2	TBL 3.3-3	F-Unit 7		3.3.5	APPLIC.		
3.3.2	ACTION	* note	01-01-A	3.3.5	ACTIONS	NOTE	
TBL 3.3-3	ACTION	15 & 16		3.3.5	ACTION	A NOTE	3.3-104
TBL 3.3-3	ACTION	15 & 16		3.3.5	ACTION	A	3.3-104
Not Used				3.3.5	ACTION	B	3.3-104
Not uSED				3.3.5	ACTION	C	3.3-104
3.3.2	TBL 4.3-2	TADOT		3.3.5	SR 3.3.5.2		
3.3.2	TBL 4.3-2	Ch Cal		3.3.5	SR3.3.5.3a		3.3-133
3.3.2	TBL 4.3-2	Ch Cal		3.3.5	SR3.3.5.3b		3.3-133
3.3.2	TBL 3.3-3	F-Unit 3c	02-05-M	3.3.6	LCO		
3.3.2	TBL 3.3-3	F-Unit 3c		3.3.6	APPLIC.		3.3-79
3.3.2	ACTION	* note	01-01-A	3.3.6	ACTIONS	NOTE	
3.3.3.1	TBL 3.3-6	ACTION37	03-14-LS29	3.3.6	ACTION	A	
3.3.2	TBL 3.3-3	APPLIC.		3.3.6	ACTION	B NOTE	
3.3.2	TBL 3.3-3	ACTION18	02-39-LG	3.3.6	ACTION	B	
3.3.2	TBL 3.3-3	APPLIC.		3.3.6	ACTION	C NOTE	
3.3.2	TBL 3.3-3	ACT. 18	02-05-M	3.3.6	ACTION	C	
3.3.2	TBL 4.3-2	Ch Ck-S	01-23-A	3.3.6	SR 3.3.6.1		
3.3.2	TBL 4.3-2	ALT-M		3.3.6	SR 3.3.6.2		

CROSS-REFERENCE TABLE FOR 3/4.3
Sorted By Improved TS

Current Specs				Improved Specs			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.3.2	TBL 4.3-2	MRT-M		3.3.6	SR 3.3.6.3		
3.3.2	TBL 4.3-2	COT-Q		3.3.6	SR 3.3.6.4		3.3-75
3.3.2	TBL 4.3-2	SRT-R		3.3.6	SR 3.3.6.5		
NA				3.3.6	SR 3.3.6.6		3.3-76
3.3.2	TBL 4.3-2	Ch Cal-R		3.3.6	SR 3.3.6.7		
3.3.2	SR	4.3.2.2	01-03-LS1	3.3.6	SR 3.3.6.8		3.3-31
3.3.2 & 3	TABLES 3.3-3,4.3-2,3.3-5,3.3-6&4.3-3			3.3.6	TBL3.3.6-1		3.3-79
NA				3.3.6	TBL3.3.6-1	FUNC. 1	3.3-77
3.3.2	TBL 4.3-2	F-U 3c1		3.3.6	TBL3.3.6-1	FUNC. 2	
3.3.3	TBL 3.3-6	INST 3a3	03-16-A	3.3.6	TBL3.3.6-1	FUNC. 3a	3.3-32
3.3.2	SR	4.3.2.2	01-02-LG	3.3.6	TBL3.3.6-1	FUNC. 3a	3.3-31
Not Used				3.3.6	TBL3.3.6-1	FUNC. 3b	3.3-32
Not Used				3.3.6	TBL3.3.6-1	FUNC. 3c	3.3-32
Not Used				3.3.6	TBL3.3.6-1	FUNC. 3d	3.3-32
3.3.2	TBL 3.3-3	FUNC. 3a		3.3.6	TBL3.3.6-1	FUNC. 4	
3.3.3.1	TBL 3.3-6	FUNC.3a3)		3.3.6	TBL3.3.6-1	NOTE (a)	3.3-31
3.3.3.1	TBL 3.3-6	FUNC.3a3)		3.3.6	TBL3.3.6-1	NOTE (b)	3.3-31
3.3.3.1	LCO			3.3.7	LCO		
3.3.3.1	APPLIC.	TBL 3.3-6		3.3.7	APPLIC.		3.3-79
3.3.3.1	ACTION	* note		3.3.7	ACTIONS	NOTE	
TBL 3.3-6	ACTION	34	03-05-LS14	3.3.7	ACTION	A	
TBL 3.3-6	ACTION	36	03-13-LS25	3.3.7	ACTION	B	3.3-51
TBL 3.3-6	ACTION	34		3.3.7	ACTION	C	
3.7.5.1	ACTION a	M 5&6		3.3.7	ACTION	D	3.3-118
CTS	ACTIONS	implicit		3.3.7	ACTION	E	
3.3.3.1	SR	4.3.3.1		3.3.7	SR	NOTE	
TBL 4.3-3	SR	Ch Ck-S		3.3.7	SR 3.3.7.1		

CROSS-REFERENCE TABLE FOR 3/4.3
Sorted By Improved TS

Current Specs				Improved Specs			
Item	Code	Para.	Change	Item	Code	Para.	Change
TBL 4.3-3	SR	CFT-Q		3.3.7	SR 3.3.7.2		3.3-75
TBL 4.3-3	SR	ALT-M		3.3.7	SR 3.3.7.3		
TBL 4.3-3	SR	MRT-M		3.3.7	SR 3.3.7.4		
TBL 4.3-3	SR	SRT-R		3.3.7	SR 3.3.7.5		
TBL 4.3-3	SR	TADOT-R		3.3.7	SR 3.3.7.6		
TBL 4.3-3	SR	Ch Cal-R		3.3.7	SR 3.3.7.7		
3.3.3.1	TABLES 3.3-6 & 4.3-3			3.3.7	TBL3.3.7-1		3.3-79
TBL 3.3-6	INST.	2a		3.3.7	TBL3.3.7-1	FUNC 1	
TBL 3.3-6	INST.	2b		3.3.7	TBL3.3.7-1	FUNC 2	
TBL 3.3-6	INST.	2c1)		3.3.7	TBL3.3.7-1	FUNC 3a	
TBL 3.3-6	INST.	2c2)		3.3.7	TBL3.3.7-1	FUNC 3b	
3.3.2	Tbl3.3.2-1	F-Unit 1&3		3.3.7	TBL3.3.7-1	FUNC 4	
3.3.3.1	LCO			3.3.8	LCO		
3.3.3.1	APPLIC	TBL 3.3-6		3.3.8	APPLIC.		
3.3.3.1	ACTION	* note		3.3.8	ACTIONS	NOTE 1	
3.3.3.1	ACTION	c	03-06-A	3.3.8	ACTIONS	NOTE 2	3.3-34
3.3.3.1	ACTIONS	30 & 32	01-43-A	3.3.8	ACTION	A	3.3-82
Not Used				3.3.8	ACTION	B	3.3-82
3.3.3.1	ACTION	30		3.3.8	ACTION	C	3.3-82
3.3.3.1	SR	4.3.3.1		3.3.8	SR	NOTE	
TBL 4.3-3	SR	Ch Ck-S		3.3.8	SR 3.3.8.1		
TBL 4.3-3	SR	CFT-Q		3.3.8	SR 3.3.8.2		3.3-75
CTS	SRS	implicit		3.3.8	SR 3.3.8.4	NOTE	
TBL 4.3-3	SR	TADOT-R "	03-08-M	3.3.8	SR 3.3.8.4		
TBL 4.3-3	SR	Ch Cal-R		3.3.8	SR 3.3.8.5		
3.3.3.1	TABLES 3.3-6 & 4.3-3			3.3.8	TBL3.3.8-1		
TBL 3.3-6	INST.	New		3.3.8	TBL3.3.8-1	FUNC 1	3.3-82

CROSS-REFERENCE TABLE FOR 3/4.3
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
TBL 3.3-6	INST.	1a1)		3.3.8	TBL3.3.8-1	FUNC 3a	3.3-82
TBL 3.3-6	INST.	1a2)		3.3.8	TBL3.3.8-1	FUNC 3b	3.3-82
TBL 3.3-6	INST.	1b		3.3.8	TBL3.3.8-1	FUNC 3c	3.3-82
TBL 3.3-6	APPLIC.			3.3.8	TBL3.3.8-1	Note (a)	
TBL 3.3-6	Note	(a)		3.3.8(new)	TBL3.3.8-1	Note (b)	3.3-82
TBL 3.3-6	Note	(b)		3.3.8(new)	TBL3.3.8-1	Note (c)	3.3-82
Not Used				3.3.9	LCO	Not Used	3.3-01

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used

Methodology for Cross-Reference Tables
(Continued)

to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another

Methodology for Cross-Reference Tables
(Continued)

document (e.g., requirements already adequately addressed by regulations).

NA This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).

Note: The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

ENCLOSURE 2
MARK-UP OF CURRENT TS

Mark-up

SPECIFICATION

PAGE

3.3.1	3/4 3-1
3.3.2	3/4 3-14
3.3.3.1	3/4 3-36
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Methodology (2 Pages)

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with ~~RESPONSE TIMES as shown in Table 3.3-2.~~

01-35-LG

APPLICABILITY: As shown in Table 3.3-1.

ACTION: *

01-01-A

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The ~~required~~ REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated ~~verified~~ to be within its limit at least once per 18 months on a ~~STAGGERED TEST BASIS~~. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

01-02-LG

01-03-LS1

~~(new) * Separate Condition entry allowed for each function.~~

01-01-A

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	REQUIRED TO TRIP	MINIMUM CHANNELS OPERABLE	CHANNELS MODES	APPLICABLE ACTION	
1. Manual Reactor Trip	2	1	2	1, 2	1	01-04-LG
	2	1	2	3*, 4*, 5*	11	01-43-A
2. Power Range, Neutron Flux						
a. High Setpoint	4	2	3	1, 2	2	
b. Low Setpoint	4	2	3	1###, 2	2 211	01-06-LS2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2 211	01-06-LS2
4. Power Range, Neutron Flux High Negative Rate	4	2	3	1, 2	2 211	01-06-LS2
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2 ⁽⁰⁾	3, 311	01-07-LS3
6. Source Range, Neutron Flux						
a. Startup	2	1	2	2##	4, 411	01-08-M
b. Shutdown	2	1	2	3*, 4*, 5*	11, 411	01-08-M
c. Shutdown	2	0	1	3 ⁽⁰⁾ , 4 ⁽⁰⁾ , and 5 ⁽⁰⁾	5	01-47-A
7. Overtemperature ΔT	4	2	3	1, 2	6 211	01-45-M
8. Overpower ΔT	4	2	3	1, 2	6 211	01-45-M
9. Pressurizer Pressure-Low	4	2	3	1 ⁽⁰⁾	6	01-19-LS8
10. Pressurizer Pressure-High	4	2	3	1, 2	6 211	01-45-M
11. Pressurizer Water Level-High	3	2	2	1 ⁽⁰⁾	6	01-19-LS8

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	REQUIRED TO TRIP	MINIMUM CHANNELS OPERABLE	CHANNELS MODES	APPLICABLE ACTION	
12. Reactor Coolant Flow-Low	3/loop			1	6	<u>01-04-LG</u>
a. Single Loop (Above P-8)	3/loop	2/loop in one loop	2/loop in each loop	1	6	<u>01-43-A</u>
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two loops	2/loop in each loop	1	6	<u>01-19-LS8</u>
13. Steam Generator Water Level Low-Low						<u>01-57-LG</u>
a. Steam Generator Water Level-Low-Low	3/S.G.	2/S.G. in one S.G.	2/S.G. in each S.G.	1.2	6	<u>01-45-M</u>
b. RCS Loop ΔT	4 (1/loop)	N.A.	N.A.	1.2	27	
DELETED						
15. Undervoltage-Reactor Coolant Pumps	2/bus	1/bus both busses	1/bus	1	28	<u>01-19-LS8</u>
16. Underfrequency-Reactor Coolant Pumps	3/bus	2 on same bus	2/bus	1	28	<u>01-50-A</u>
17. Turbine Trip						<u>01-19-LS8</u>
a. Low Autostop Oil Pressure	3	2	2	1	7	<u>01-50-A</u>
b. Turbine Stop Valve Closure	4	4	4	1	7	<u>01-48-LS4</u>
						<u>01-48-LS4</u>

TABLE 3.3-1 (Continued)

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	REACTOR TRIP SYSTEM INSTRUMENTATION		CHANNELS MODES	APPLICABLE ACTION	
		REQUIRED TO TRIP	MINIMUM CHANNELS OPERABLE			
18. Safety Injection Input from ESF	2	1	2	1, 2	26	01-04-LG 01-43-A
19. Reactor Coolant Pump Breaker Position Trip above P-7	1/breaker	2	1/breaker	1 ³⁰	9 6	01-49-LS18
20. Reactor Trip Breakers	2 2	1 1	2 2	1, 2 3*, 4*, 5*	10, 12 11	01-14-A
(new) Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1 each per RTB 1 each per RTB			1, 2 3*, 4*, 5*	12 11	01-14-A 01-14-A
21. Automatic Trip and Interlock Logic	2 2	1 1	2 2	1, 2 3*, 4*, 5*	26 11	
22. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8	01-51-LG 01-12-M
b. Low Power Reactor Trips Block, P-7 P-10 Input P-13 Input	4 1 per train 2	2 1	3 2	1 1	8# 8 1 8#	01-05-A 01-12-M
c. Power Range Neutron Flux, P-8	43	2	3	1	8# 8 1	01-37-A 01-05-A
d. Power Range Neutron Flux, P-9	43	2	3	1	8# 8 1	01-37-A 01-12-M
e. Power Range Neutron Flux, P-10	43	2	3	1, 2	8#	01-05-A 01-37-A
f. Turbine Impulse Chamber Pressure, P-13 (Input to P-7)	2	1	2	1	8# 8 1	01-05-A 01-12-M
23. Seismic Trip	3 direc- tions (x,y,z) in 3 locations	2/3 loca- tions one direction	2/3 loca- tions all directions	1,2	13	

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

* When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal or all rods are not fully inserted.	01-55-LS39
#The provisions of Specification 3.0.4 are not applicable.	01-05-A
##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.	
###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.	
(new) ** Not required to be performed until 12 hours after input from one Power Range Neutron Flux channel to QPTR is inoperable and THERMAL POWER is > 75% RTP.	01-53-A
(new) (f) With the RTB's open or the Rod Control System incapable of withdrawal. In this condition, source range function does not provide reactor trip but does provide indication.	01-47-A
(new) (j) Above the P-9 (Power Range Neutron Flux) Interlock.	01-48-LS4
(new) (k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.	01-14-A
(new) (d) Above the P-6 (Intermediate Range Neutron Flux) Interlock.	01-07-LS3
(new) (g) Above the P-7 (Low Power Reactor Trips Block) Interlock.	01-19-LS8

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than the Minimum REQUIRED Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours.	01-04-LG
ACTION 2 - With the number of OPERABLE channels one less than the Total Number of REQUIRED Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:	01-43-A
a. The inoperable channel is placed in the tripped condition within 6 hours.	
b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1 and set point adjustment, and	01-04-LG 01-17-A
c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 (2) hours; or, if the power range neutron flux input to the QUADRANT POWER TILT RATIO is inoperable, the QPTR is monitored per Specification 4.2.4.2*** when THERMAL POWER is greater than or equal to 50% of RATED THERMAL POWER.	01-18-LS7 01-53-A 01-56-A
(new) Otherwise be in MODE 3 within 12 hours.	01-18-LS7
(new) ACTION 2.1 With one Channel Inoperable, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied (Note: The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1 and setpoint adjustment):	01-06-LS2
a. Place the inoperable channel in tripped condition within 6 hours, or	
b. Be in MODE 3 within 12 hours.	01-17-A

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 -	With the number of channels OPERABLE one less than the Minimum REQUIRED Channels OPERABLE requirement and with the THERMAL POWER level:	01-04-LG
a.	Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and	01-07-LS3
b.	Above the P-6 Setpoint, but below 10% of RATED THERMAL POWER, within 24 hours restore the inoperable channel to OPERABLE status prior to increasing or reduce Thermal Power to less than P-6, or increase THERMAL POWER above 10% of RATED THERMAL POWER.	01-07-LS3
ACTION 3.1 (new)	With two Intermediate Range Neutron Flux Channels inoperable and with the THERMAL POWER level above the P-6 Setpoint but below 10% of RATED THERMAL POWER, immediately suspend operations involving positive reactivity additions and, within 2 hours, reduce THERMAL POWER to less than the P-6 Setpoint.	01-07-LS3
ACTION 4 -	With the number of channels OPERABLE one less than the Minimum REQUIRED Channels OPERABLE requirement immediately suspend all operations involving positive reactivity changes.	01-04-LG 01-08-M
ACTION 4.1 (new)	With no source range neutron flux channels OPERABLE immediately open the reactor trip breakers	01-08-M
ACTION 5 -	With the number of channels OPERABLE one less than the Minimum REQUIRED Channels OPERABLE requirement, immediately suspend operations involving positive reactivity additions and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.	01-04-LG 01-09-M
ACTION 6 -	With the number of OPERABLE channels one less than the Total Number of REQUIRED Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:	01-43-A
a.	The inoperable channel is placed in the tripped condition within 6 hours, and or	01-19-LS8
b.	Reduce Thermal Power to < P-7 within 12 hours. The Minimum Channels OPERABLE requirement is met; however,	01-04-LG
	Note: The inoperable channel or one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.	
ACTION 7 -	With the number of OPERABLE channels less than the Total Number of Required Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel(s) is (are) placed in the tripped condition within 6 hours or THERMAL POWER is decreased < P-9 in 10 hours.	01-43-A 01-48-LS4
	NOTE: The inoperable low Autostop oil pressure channel may be bypassed for up to 4 hours for surveillance testing of other channels.	
ACTION 8 -	With less than the Minimum Number of Required Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3 be in at least HOT STANDBY within 7 hours.	01-04-LG 01-52-LG 01-12-M
ACTION 8.1 (new)	With one or more Required Channels inoperable, within 1 hour verify the interlock is in its required state for the existing plant condition or be in at least MODE 2 within 7 hours.	01-12-M

TABLE 3.3-1 (Continued)
ACTION STATEMENTS (Continued)

ACTION 9	With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within the next 6 hours.	01-49-LS18
ACTION 10	With the number of channel trains OPERABLE one less than the Minimum Total Number of Required Channels, OPERABLE requirement, restore the inoperable train to operable status within 1 hour or be in at least HOT STANDBY within 6 hours; however, one channel train may be bypassed for up to 2 hours for maintenance or surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.	01-04-LG 01-13-LS6
ACTION 11	With the number of OPERABLE channels or trains one less than the Minimum Required Channels or trains, OPERABLE requirement, restore the inoperable channel or train to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour, fully insert all rods and place the Rod control System in a condition incapable of rod withdrawal.	01-04-LG 01-55-LS39
ACTION 12	With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 10 or be in at least HOT STANDBY within the next 6 hours. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.	01-14-A
ACTION 13	With the number of OPERABLE channels one less than the Total Number of Required Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:	01-43-A
	a. The Minimum Channels OPERABLE requirement is met, and	01-04-LG
	b. The inoperable channel is placed in the tripped conditions within 6 hours; however, the inoperable channel may be bypassed for up to 72 hours for surveillance testing per Specification 4.3.1.1 or for performing maintenance.	
ACTION 26	With the number of OPERABLE channels one less than the Minimum Required Channels, OPERABLE requirement, restore the inoperable Channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.	01-04-LG
ACTION 27	With the number of OPERABLE channels less than the Total Number of Required Channels, STARTUP and/or POWER OPERATION may proceed provided that within 6 hours, for the affected RCS Loop Delta-T channel(s), either:	01-43-A
	a. The Trip Time Delay threshold power level for zero seconds time delay is adjusted to 0% RTP, or	01-43-A
	b. With the number of OPERABLE channels one less than the Total Number of Required Channels, the affected Steam Generator Water Level-Low-Low channels are placed in the tripped condition.	
ACTION 28	With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:	01-50-A
	a. The inoperable channel is placed in the trip condition within 6 hours, and	
	b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.	

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

<u>FUNCTIONAL UNIT</u>	<u>RESPONSE TIME</u>	<u>01-35-LG</u>
1. Manual Reactor Trip	N.A.	
2. Power Range, Neutron Flux	≤ 0.5 second (1)	
3. Power Range, Neutron Flux, High Positive Rate	N.A.	
4. Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second (1)	
5. Intermediate Range, Neutron Flux	N.A.	
6. Source Range, Neutron Flux	≤ 0.5 second (1)	
7. Overtemperature ΔT	≤ 7 seconds (1)	
8. Overpower ΔT	≤ 7 seconds (1)	
9. Pressurizer Pressure-Low	≤ 2 seconds	
10. Pressurizer Pressure-High	≤ 2 seconds	
11. Pressurizer Water Level-High	N.A.	
12. Reactor Coolant Flow-Low		
a. Single Loop (Above P-8)	≤ 1 second	
b. Two Loops (Above P-7 and below P-8)	≤ 1 second	
13. Steam Generator Water Level-Low-Low		
a. Steam Generator Water Level-Low-Low	≤ 2 seconds (2)	
b. RCS Loop ΔT Equivalent Power	N.A.	
14. DELETED		
15. Undervoltage-Reactor Coolant Pumps	≤ 1.2 seconds	
16. Underfrequency-Reactor Coolant Pumps	< 0.6 second	

(1) Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input to the first electronic component in channel. 01-41-A

(2) Does not include Trip Time Delays. Response times include the transmitters, Eagle-21 Process Protection cabinets, Solid State Protection System cabinets and actuation devices only. This reflects the response times necessary for THERMAL POWER in excess of 50% RTP. 01-35-LG

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

<u>FUNCTIONAL UNIT</u>	<u>RESPONSE TIME</u>	<u>01-35-LG</u>
17.Turbine Trip		
a. Low Fluid Oil Pressure	N.A.	
b. Turbine Stop Valve	N.A.	
18.Safety Injection Input from ESF	N.A.	
19.Reactor Coolant Pump Breaker Position Trip	N.A.	
20.Reactor Trip Breakers	N.A.	
21Automatic Trip and Interlock Logic	N.A.	
22.Reactor Trip System Interlocks	N.A.	
23.Seismic Trip	N.A.	

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3*, 4*, 5*	01-44-A 01-32-LG
2. Power Range, Neutron Flux a. High Setpoint	S	D(2-4), M(3-4), Q(4-6), R(4, 5, 22) R(4, 5, 22)	Q	N.A.	N.A.	1, 2	01-21-A 01-23-A 01-22-M 01-23-A
b. Low Setpoint	S	R(4, 5, 22) R(4, 5, 22)	S/U(1, 20) Q(19, 20) N.A.	N.A.	N.A.	1###, 2	01-39-A 01-23-A 01-39-A 01-23-A
3. Power Range, Neutron Flux, High Positive Rate	N.A. R(4, 5, 22)	Q	N.A.	N.A.	1, 2		01-39-A 01-23-A
4. Power Range, Neutron Flux, High Negative Rate	N.A. R(4, 5, 22)	Q	N.A.	N.A.	1, 2		01-39-A 01-23-A
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1, 20) Q(19, 20)	N.A.	N.A.	1###, 2	01-22-M 01-26-LG
6. Source Range, Neutron Flux	S R(4, 5)	S/U(1, 8), Q(8, 19)		N.A.	N.A.	2##, 3, 4, 5	01-28-A 01-27-LS10
7. Overtemperature ΔT	S	R(22), M(3, 4), Q(4, 6) R(22)	Q	N.A.	N.A.	1, 2	01-23-A 01-21-A 01-23-A
8. Overpower ΔT	S	R(22)	Q	N.A.	N.A.	1, 2	01-23-A
9. Pressurizer Pressure-Low	S	R(22)	Q	N.A.	N.A.	1	01-23-A
10. Pressurizer Pressure-High	S	R(22)	Q	N.A.	N.A.	1, 2	01-23-A
11. Pressurizer Water Level-High	S	R(22)	Q	N.A.	N.A.	1	01-23-A
12. Reactor Coolant Flow-Low	S	R(22)	Q	N.A.	N.A.	1	01-23-A

TABLE 4.3-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>	<u>01-44-A</u>
13. Steam Generator Water Level-Low-Low							
a. Steam Generator Water Level-Low-Low	S	R(22) Q	Q	N.A.	N.A.	1, 2	01-23-A
b. RCS Loop ΔT	N.A.	R (22) Q	Q	N.A.	N.A.	1, 2	01-23-A
14. DELETED							
15. Undervoltage-Reactor Coolant Pumps	N.A.	R (22) N.A.	N.A.	Q(9) N.A.	N.A.	±	01-23-A 01-16-LS40
16. Underfrequency-Reactor Coolant Pumps	N.A.	R (22) N.A.	N.A.	Q(9) N.A.	N.A.	±	01-23-A 01-16-LS40
17. Turbine Trip							
a. Low Fluid Oil Pressure	N.A.	R(22) N.A.	N.A.	S/U(1, 9)	N.A.	±	01-36-M
b. Turbine Stop Valve Closure	N.A.	R(22) N.A.	N.A.	S/U(1, 9)	N.A.	±	01-23-A 01-24-LS9
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2	
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	R	N.A.	±	
20. Reactor Trip System Interlocks							
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2##	
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	±	
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	±	01-51-LG

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>	<u>01-44-A</u>
20. Reactor Trip System Interlocks (Continued)							
d. Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1	
e. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2	
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R (22)	R	N.A.	N.A.	1	01-23-A
21. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 10)	N.A.	1, 2, 3*, 4*, 5*	01-32-LG
(new) Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	N.A.	N.A.	N.A.	M(7)	N.A.		01-14-A
22. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*	
23. Seismic Trip	N.A.	R	N.A.	R	R	1, 2	
24. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(7, 15), R(16)	N.A.	1, 2, 3*, 4*, 5*	01-32-LG

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*	- When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal or all rods not fully inserted.	<u>01-55-LS39</u>
##	- Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.	
###	- Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.	
(1)	- If not performed in previous 31 <u>92</u> days.	<u>01-24-LS9</u>
(1a)	If not performed in previous 31 days	<u>01-24-LS9</u>
(2)	- Heat balance only, above 15% of RATED THERMAL POWER. During startup in MODE 1 above 15% of RATED THERMAL POWER, the required heat balance shall be performed prior to exceeding 30% of RATED THERMAL POWER, or within 24 hours, whichever occurs first. Adjust channel if absolute difference greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.	<u>01-25-A</u>
(3)	- Compare incore to excore axial flux difference above within 24 hours after Thermal Power is greater than or equal to 1550% of RATED THERMAL POWER and at least once per 31 Effective Full Power days. Re-Calibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.	<u>01-25-A</u>
(4)	- Neutron detectors may be excluded from CHANNEL CALIBRATION.	
(5)	- Detector plateau curves shall be obtained and evaluated for the source range neutron flux channels. For the Intermediate Range and Power Range Neutron Flux channels a test shall be performed that shows allowed variances of detector voltage do not effect detector operation. For the Intermediate Range and Power Range Neutron Flux Channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.	<u>01-26-LG</u>
(6)	- Incore - Excore Calibration, above within 24 hours after Thermal Power is a 75% of RATED THERMAL POWER and at least once per 92 Effective Full Power days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.	<u>01-25-A</u>
(7)	- Each train shall be tested at least every 62 <u>31</u> days on a STAGGERED TEST BASIS.	
(8)	- Quarterly Surveillances in MODES 3*, 4* and 5* performed quarterly and prior to startup shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.	<u>01-28-A</u>
(9)	- Setpoint verification is not applicable.	
(10)	The TRIP ACTUATING DEVICE OPERATIONAL TEST shall separately verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.	<u>01-32-LG</u>
(11)	- Deleted	
(12)	- Deleted	
(13)	- Deleted	

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

(14) —	The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).	<u>01-32-LG</u>
(15) -	Test local manual shunt trip prior to placing breaker in service.	
(16) —	Test automatic undervoltage trip.	<u>01-32-LG</u>
(19) —	The CHANNEL OPERATIONAL TEST shall be performed within 12 hours after reducing power below P-10 for the power range and intermediate range instrumentation and within 4 hours after reducing power below P-6 for the source range instrumentation, if not performed within the previous 92 days. With the Rod Control System capable of rod withdrawal, or all rods not fully inserted, the COT is not required prior to entering MODE 3 from MODE 2 until 4 hours after entering MODE 3.	<u>01-22-M</u>
(20) —	Surveillance shall also include verification that permissives P-6 (new) and P-10 are in their required state for existing plant conditions.	<u>01-22-M</u>
(22) — (new)	Includes verification of time constants.	<u>01-23-A</u>

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION: ✖

- a. With an ESFAS Instrumentation Channel or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Values column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.
- b. With an ESFAS Instrumentation Channel or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated/verified OPERABLE by the performance of the Engineered Safety Feature Actuation System Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The required ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated/verified to be within the limit at least once per 18 months on a STAGGERED TEST BASIS**. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

(new) * Separate ACTION entry is allowed for each Functional Unit.

(new) ** Not required to be performed for the turbine driven auxiliary feedwater pump until 24 hours after steam generator pressure ≥650 psig.

02-01-A

01-35-LG

01-01-A

02-04-LG

02-04-LG

01-03-LS1

01-02-LG

01-03-LS1

02-40-A

01-01-A

02-40-A

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF REQUIRED CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>	<u>01-04-LG</u> <u>01-43-A</u>
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Fan Cooler Units, and Component Cooling Water)						<u>02-19-LG</u>
a. Manual Initiation	2	1	2	1, 2, 3, 4	19	
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14	
c. Containment Pressure-High	3	2	2	1, 2, 3, 4	20	<u>02-08-M</u>
d. Pressurizer Pressure-Low	4	2	3	1, 2, 3#	20	
e. DELETED						
f. Steam Line Pressure- Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3#	20	

Table 3.3-3 (continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF REQUIRED CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>	<u>01-04-LG</u> <u>01-43-A</u> <u>02-28-LG</u>
2. Containment Spray (Coincident with SI Signal)						
a. Manual	2	2 with 2 coincident switches	2	1, 2, 3, 4	19	
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14	
c. Containment Pressure-High-High	4	2	3	1, 2, 3, 4	17	
3. Containment Isolation						
a. Phase "A" Isolation						
1) Manual	2	1	2	1, 2, 3, 4	19	
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14	
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.					
b. Phase "B" Isolation						
1) Manual	2	2 with 2 coincident switches	2	1, 2, 3, 4	19	

TABLE 3.3-3 (Continued)

FUNCTIONAL UNIT	ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION		MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	01-43-A
	TOTAL NO. OF CHANNELS REQUIRED	CHANNELS TO TRIP				01-04-LG
3. Containment Isolation (Continued)						
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14	
3) Containment Pressure-High-High	4	2	3	1, 2, 3, 4	17	
c. Containment Ventilation Isolation						
1) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4 during CORE ALTERATIONS, during movement of irradiated fuel assemblies within containment.	18, 37	03-14-LS29
2) Deleted						
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.					
4) Containment Ventilation Exhaust Radiation-High (RM-44A and 44B)	2	1	2	1, 2, 3, 4.	18, 37	03-14-LS29
4. Steam Line Isolation						
a. Manual	1 manual switch/steam line	1 manual switch/steam line	1 manual switch/operating steam line	1, 2 ^(a) , 3 ^(a) , 4	24	02-07-LS11 02-38-LS35

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF REQUIRED CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	01-43-A	01-04-LG
4. Steam Line Isolation (Continued)							
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2 ^(a) , 3 ^(a)	22	02-07-LS11	
c. Containment Pressure-High-High	4	2	3	1, 2 ^(a) , 3 ^(a)	17 ⁽¹⁾	02-15-M	02-07-LS11
d. Steam Line Pressure-Low	3/steam line	2/steam line in any steam line	2/steam line	1, 2 ^(a) , 3 ^(a)	20	02-07-LS11	
e. Negative Steam Line Pressure Rate - High		3/steam line	2/steam line in any steam line	2/steam line	3 ^(a) 20	02-07-LS11	
5. Turbine Trip & Feedwater Isolation							
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2 ^(b)	25	02-07-LS11	
b. Steam Generator Water Level-High-High	3/stm. gen.	2/stm. gen. in any operating stm. gen.	2/stm. gen.	1, 2 ^(b) in each operating stm. gen.	20 35 ⁽²⁾	02-07-LS11	02-08-M

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF REQUIRED CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS	APPLICABLE MODES OPERABLE	ACTION	01-04-LG
						01-43-A
6. Auxiliary Feedwater						
a. Manual Initiation	1 manual switch/pump	1 manual switch/pump	1 manual switch/pump	1, 2, 3	24	
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22	
c. Stm. Gen. Water Level-Low-Low						
1) Start Motor Driven Pumps						<u>02-09-LG</u>
a. Steam Generator Water Level-Low-Low	3/S.G.	2/S.G. in one S.G.	2/S.G. in each S.G.	1, 2, 3###	20	
b. RCS Loop ΔT	4 (1/loop)	N.A.	N.A.	1, 2	29	
2) Start Turbine Driven Pump						<u>02-09-LG</u>
a. Steam Generator Water Level-Low-Low	3/S.G.	2/S.G. in any 2 S.G.	2/S.G. in each S.G.	1, 2, 3###	20	
b. RCS loop ΔT	4 (1/loop)	N.A.	N.A.	1, 2	29	
d. Undervoltage-RCP Bus Start Turbine-Driven Pump	2/bus	1/bus on both busses	1/bus	1	35	<u>02-19-LG</u>
e. Safety Injection Start Motor Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.					<u>02-19-LG</u>

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF REQUIRED CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	01-04-LG
						01-43-A
7. Loss of Power (4.16 kV Emergency Bus Undervoltage)						02-11-A
a. First Level				1, 2, 3, 4**		02-36-M
1) Diesel Start	1/Bus	1/Bus	1/Bus		16	
2) Initiation of Load Shed	2/Bus	2/Bus	2/Bus		1615	02-48-LS2
b. Second Level				1, 2, 3, 4**		02-36-M
1) Undervoltage Relays	2/Bus	2/Bus	2/Bus		1615	02-48-LS2
2) Timers to Start Diesel	1/Bus	1/Bus	1/Bus		16	
3) Timers to Shed Load	1/Bus	1/Bus	1/Bus		16	
8. Engineered Safety Features Actuation System Interlocks						
a. Pressurizer Pressure, P-11	2	2	2	1, 2, 3	21	01-37-A
b. DELETED						
c. Reactor Trip, P-4	2	2	2	1, 2, 3	23	
9. Residual Heat Removal pump trip-low RWST level	3			1, 2, 3, 4	20	02-29-M

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

- # Trip function may be blocked in this MODE below the P11 (Pressurizer Pressure Interlock) Setpoint.
- ## Trip function automatically blocked above P-11 (Pressurizer Pressure Interlock) Setpoint and is automatically blocked below P-11 when Safety Injection on Steam Line Pressure-Low is not blocked.
- ### For Mode 3, the Trip Time Delay associated with the Steam Generator Water Level-Low-Low channel must be less than or equal to 464.1 seconds.

02-36-M

(new) ** When associated DG is required to be OPERABLE by LCO 3.8.1.2, "AC Sources Shutdown."

(new) (a) Not applicable when all MSIVs are closed and deactivated

02-07-LS11

(new) (b) Not applicable when all MFIVs, main feedwater regulating valves, and main feedwater regulating bypass valves are closed and deactivated or isolated by a closed manual isolation valve.

02-07-LS11

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Required Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

01-04-LG

~~ACTION 15 - With the number of OPERABLE Channels less than the Required Channels, declare the affected emergency Diesel Generator(s) inoperable and comply with the ACTION statements of Specification 3.8.1.1; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.~~

02-48-LS28

01-43-A

ACTION 16 - With the number of OPERABLE Channels one less than the Total Number of Required Channels, declare the affected Emergency Diesel Generator(s) inoperable and comply with the ACTION statements of Specification 3.8.1.1; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

01-43-A

02-11-A

~~ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels one containment pressure channel inoperable, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met within 6 hours, or be in MODE 3 in 12 hours and in MODE 5 in 42 hours.~~

01-43-A

02-15-M

01-04-LG

Note: one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

~~(new) ACTION 17.1 - With one containment pressure channel inoperable, operation may proceed provided the inoperable channel is placed in the bypassed condition within 6 hours, or be in MODE 3 in 12 hours and in MODE 4 in 18 hours. Note: one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.~~

02-15-M

ACTION 18 - With less than the Minimum Required Channels OPERABLE requirement, comply with ACTION 37, and operation may continue beyond the 4 hour period provided the containment purge supply and exhaust valves (RCV 11, 12, FGV 660, 661, 662, 663, 664) are maintained closed.

01-04-LG

02-05-M

02-39-LG

03-14-LS29

ACTION 19 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement one channel inoperable, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

01-04-LG

ACTION 20 - With the number of OPERABLE channels one less than the Total Number of Channels one channel inoperable, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

02-08-M

01-43-A

- a. The inoperable channel is placed in the tripped condition within 6 hours, and or
- b. Be in MODE 3 in 12 hours and in MODE 4 in 18 hours. NOTE: The Minimum Channels OPERABLE requirement is met; however, the inoperable channel or one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

01-04-LG

ACTION 20.1 - With one channel inoperable, STARTUP and/or POWER OPERATION may proceed (new) provided the following conditions are satisfied:

02-29-M

- a. The inoperable channel is placed in the bypassed condition within 6 hours, and the inoperable channel is returned to an OPERABLE status within 72 hours, or
- b. Immediately enter 3.0.3.

NOTE: One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1

ACTION 20.2 - With one channel inoperable, STARTUP and/or POWER OPERATION may (new) proceed provided the following conditions are satisfied:

02-08-M

- a. The inoperable channel is placed in the tripped condition within 6 hours, or
- b. Be in MODE 3 in 12 hours and in MODE 5 in 42 hours. NOTE: The inoperable channel or one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 21 - With the number of OPERABLE channels less than the Minimum Number of Required Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3 be in MODE 3 in 7 hours and MODE 4 in 13 hours.	<u>01-04-LG</u> <u>02-14-M</u> <u>01-52-LG</u>
ACTION 22 - With the number of OPERABLE Channels one less than the Minimum Required Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.	<u>01-04-LG</u>
ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Required Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.	<u>01-43-A</u>
ACTION 24 - With the number of OPERABLE channels one less than the Total Number of Required Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated pump or valve inoperable and take the ACTION required by Specification 3.7.1.5 or 3.7.1.2 as applicable.	<u>01-43-A</u>
ACTION 25 - With the number of OPERABLE channels one less than the Minimum Required Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.	<u>01-04-LG</u>
ACTION 29 - With the number of OPERABLE channels less than the Total Number of Required Channels, STARTUP and/or POWER OPERATION may proceed provided that within 6 hours, for the affected RCS Loop Delta-T channel(s), either:	<u>01-43-A</u>
a. The Trip Time Delay threshold power level for zero seconds time delay is adjusted to 0% RTP, or	
b. With the number of OPERABLE channels one less than the Total Number of Required Channels, the affected Steam Generator Water Level-Low-Low channels are placed in the tripped condition.	<u>01-43-A</u>
ACTION 35 - With the number of OPERABLE channels one less than the Total Number of Required Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:	<u>01-43-A</u>
a. The inoperable channel is placed in the trip condition within 6 hours, and or be in MODE 2 in 12 hours	<u>02-08-M</u>
b. The Minimum Channels OPERABLE requirement is met; however, The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.	<u>01-04-LG</u>
ACTION 35.2 - With the number of OPERABLE channels one less than the Required Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:	<u>02-08-M</u>

a. The inoperable channel is placed in the trip condition within 6 hours or

b. Be in MODE 2 in 6 hours for Function 6.g. or be in MODE 3 in 12 hours for Function 5.b.

NOTE: The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT

TRIP SETPOINT

ALLOWABLE VALUES

1. ~~Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Fan Cooler Units, and Component Cooling Water)~~

02-19-LG

a. Manual Initiation

N.A.

N.A

b. Automatic Actuation Logic and Actuation Relays

N.A.

N.A

c. Containment Pressure-High

≤ 3 psig

≤ 3.3 psig

d. Pressurizer Pressure-Low

≥ 1850 psig

≥ 1844.4 psig

e. DELETED

f. Steam Line Pressure-Low

≥ 600 psig (Note 1)

≥ 594.6 psig (Note 1)

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
2. Containment Spray (Containment with SI Signal)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	≤ 22 psig	≤ 22.3 psig
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
b. Phase "B" Isolation		
1) Manual	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure-High-High	≤ 22 psig	≤ 22.3 psig

02-28-LG

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
3. Containment Isolation (Continued)		
c. Containment Ventilation Isolation		
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
2) Deleted		
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
4) Containment Ventilation Exhaust Radiation-High (RM-44A and 44B)	Per the ODCP	
4. Steam Line Isolation		
a. Manual	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	≤ 22 psig	≤ 22.3 psig
d. Steam Line Pressure-Low	≥ 600 psig (Note 1)	≥ 594.6 psig (Note 1)

02-05-M

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
e. Negative Steam Line Pressure Rate - High	≤ 100 psi (Note 3)	≤ 105.4 psi (Note 3)
5. Turbine Trip and Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water level-High-High	$\leq 75\%$ of narrow range instrument span each steam generator.	$\leq 75.5\%$ of narrow range instrument span each steam generator.
6. Auxiliary Feedwater		
a. Manual	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Steam Generator Water Level-Low-Low	$\geq 7.2\%$ of narrow range instrument span each steam generator.	$\geq 6.8\%$ of narrow range instrument span each steam generator.
Coincident with:		
1) RCS Loop ΔT Equivalent to Power $\leq 50\%$ RTP With a time delay (TD)	RCS Loop ΔT variable input $\leq 50\%$ RTP \leq TD (Note 2)	RCS Loop ΔT variable input $\leq 51.5\%$ RTP $\leq (1.01)TD$ (Note 2)
Or		
2) RCS Loop ΔT Equivalent to Power $>50\%$ RTP With no time delay	RCS Loop ΔT variable input $> 50\%$ RTP TD = 0	RCS Loop ΔT variable input $> 51.5\%$ RTP TD = 0
d. Undervoltage - RCP	≥ 8050 volts	≥ 7730 volts
e. Safety Injection		See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.

TABLE 3.3-4 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>	<u>02-11-A</u>
7. Loss of Power (4.16 kV Emergency Bus Undervoltage)			
a. First Level			
1) Diesel Start	< 0.8 second time delay and > 2583 volts with a 10 second time delay One relay	> 0 volts with a < 0.8 second time delay and > 2583 volts with < 10 second time delay One relay	
2) Initiation of Load Shed	< 0 volts with a 4 second time delay and > 2583 volts with a 25 second time delay with one relay ≥ 2870 volts, instantaneous	> 0 volts with a < 4 second time delay and > 2583 volts with a < 25 second time delay with one relay ≥ 2870 volts, instantaneous	
b. Second Level			
1) Diesel Start			
2) Initiation of Load Shed	< 10 second time delay > 3785 volts with a < 20 second time delay	> 3785 volts with a < 10 second time delay > 3785 volts with a < 20 second time delay	
8. Engineered Safety Features Actuation System Interlocks			
a. Pressurizer Pressure, P-11	≤ 1915 psig	≤ 1920.6 psig	
b. DELETED			
c. Reactor Trip, P-4	N.A.	N.A.	
9. Residual Heat Removal pump trip - low RWSI level	≤ 33% level	≤ 32.9% level	<u>02-29-M</u>

NOTE 1: Time constants utilized in the lead-lag controller for Steam Pressure - Low are $\tau_1 = 50$ seconds and $\tau_2 = 5$ seconds.

NOTE 2: Steam Generator Water Level Low-Low Trip Time Delay

$$TD = B1(P)^3 + B2(P)^2 + B3(P) + B4$$

Where: P = RCS Loop ΔT Equivalent to Power (%RTP), P ≤ 50% RTP

TD = Time delay for Steam Generator Water Level Low-Low (in seconds)

$$\begin{aligned} B1 &= -0.007128 \\ B2 &= +0.8099 \\ B3 &= -31.40 \\ B4 &= +464.1 \end{aligned}$$

NOTE 3: Time constants utilized in the rate-lag controller for Negative Steam Line Pressure Rate-High are $\tau_3 = 50$ seconds and $\tau_4 = 50$ seconds.

TABLE 3.3-5

01-35-LG

ENGINEERED SAFETY FEATURES RESPONSE TIMESINITIATING SIGNAL AND FUNCTIONRESPONSE TIME IN SECONDS

1. Manual Initiation

a. Safety Injection (ECCS)

N.A.

1) Feedwater Isolation

N.A.

2) Reactor Trip

N.A.

3) Phase "A" Isolation

N.A.

4) Containment Ventilation Isolation

N.A.

5) Auxiliary Feedwater

N.A.

6) Component Cooling Water

N.A.

7) Containment Fan Cooler Units

N.A.

8) Auxiliary Saltwater Pumps

N.A.

b. Phase "B" Isolation

1) Containment Spray (Coincident with
SI Signal)

N.A.

2) Containment Ventilation Isolation

N.A.

c. Phase "A" Isolation

1) Containment Ventilation Isolation

N.A.

d. Steam Line Isolation

N.A.

2. Containment Pressure-High

a. Safety Injection (ECCS)

 $\leq 27(7)/25(4)$

1) Reactor Trip

 ≤ 2

2) Feedwater Isolation

 ≤ 63

3) Phase "A" Isolation

 $\leq 18(1)/28(3)$

4) Containment Ventilation Isolation

N.A.

5) Auxiliary Feedwater

 $\leq 60(3)$

6) Component Cooling Water

 $\leq 38(1)/48(3)$

7) Containment Fan Cooler Units

 $\leq 40(3)$

8) Auxiliary Saltwater Pumps

 $\leq 48(1)/58(3)$

3. Pressurizer Pressure-Low

a. Safety Injection (ECCS)

 $\leq 27(7)/25(4)/35(5)$

1) Reactor Trip

 ≤ 2

2) Feedwater Isolation

 ≤ 63

3) Phase "A" Isolation

 $\leq 18(1)$

4) Containment Ventilation Isolation

N.A.

5) Auxiliary Feedwater

 $\leq 60(3)$

6) Component Cooling Water

 $\leq 48(3)/38(1)$

7) Containment Fan Cooler Units

 $\leq 40(3)$

8) Auxiliary Saltwater Pumps

 $\leq 58(3)/48(1)$

TABLE 3.3-5 (Continued)

01-35-LG

ENGINEERED SAFETY FEATURES RESPONSE TIMES

<u>INITIATING SIGNAL AND FUNCTION</u>	<u>RESPONSE TIME IN SECONDS</u>
4. Negative Steam Line Pressure Rate-High	
a. Steam Line Isolation	≤ 8
5. DELETED	
6. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 25(4)/35(5)
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 63
3) Phase "A" Isolation	≤ 18(1)/28(3)
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	≤ 60(3)
6) Component Cooling Water	≤ 38(1)/48(3)
7) Containment Fan Cooler Units	≤ 40(3)
8) Auxiliary Saltwater Pumps	≤ 48(1)/58(3)
b. Steam Line Isolation	≤ 8

TABLE 3.3-5 (Continued)

01-35-LG

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Containment Pressure-High-High	
a. Containment Spray	≤ 48.5(6)
b. Phase "B" Isolation	N.A.
c. Steam Line Isolation	≤ 7
8. Steam Generator Water Level-High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 66
9. Steam Generator Water Level Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	≤ 60(3)(8)
b. Turbine-Driven Auxiliary Feedwater Pump	≤ 60(8)
10. RCP Bus Undervoltage	
Turbine-Driven Auxiliary Feedwater Pump	≤ 60
11. Deleted	
12. Containment Ventilation Exhaust Radiation-High	
Containment Ventilation Isolation	≤ 11

TABLE NOTATIONS

- (1) Diesel generator starting delay not included because offsite power available.
- (2) Notation deleted.
- (3) Diesel generator starting and loading delays included.
- (4) Diesel generator starting delay not included because offsite power is available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps (where applicable). Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.
- (5) Diesel generator starting and sequence loading delays included. Offsite power is not available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.
- (6) The maximum response time of 48.5 seconds is the time from when the containment pressure exceeds the High-High Setpoint until the spray pump is started and the discharge valve travels to the fully open position assuming off-site power is not available. The time of 48.5 seconds includes the 28-second maximum delay related to ESF loading sequence. Spray riser piping fill time is not included. The 80-second maximum spray delay time does not include the time from LOCA start to "P" signal.
- (7) Diesel generator starting and sequence loading delays included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included. Response time limit includes opening of valves to establish SI flow path and attainment of discharge pressure for centrifugal charging pumps, SI, and RHR pumps (where applicable).
- (8) Does not include Trip Time Delays. Response times include the transmitters, Eagle 21 Process Protection cabinets, Solid State Protection System cabinets and actuation devices only. This reflects the response times necessary for THERMAL POWER in excess of 50% RTP.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

01-44-A

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALI- BRATION	CHANNEL OPERA- TIONAL TEST	TRIP ACTUATING DEVICE OPERA- TIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
1. Safety Injection, (Reactor Trip Feedwater Isolation, Start Diesel Generators, Containment Fan Cooler Units, and Component Cooling Water)									02-19-LG
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4	
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3, 4	
4 c. Containment Pressure- High	S	R(6) Q		N.A.	N.A.	N.A.	N.A.	1, 2, 3,	01-23-A
d. Pressurizer Pressure-Low	S	R(6) Q		N.A.	N.A.	N.A.	N.A.	1, 2, 3	01-23-A
e. DELETED									01-23-A
f. Steam Line Pressure-Low	S	R(6) Q		N.A.	N.A.	N.A.	N.A.	1, 2, 3	01-23-A
2. Containment Spray (coincident with SI signal)									02-28-
4 a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3,	
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3, 4	
1, 2, 3, 4** c. Containment Pressure- High-High	S	R(6) Q		N.A.	N.A.	N.A.	N.A.		01-23-A

* These changes from License Amendments 84 & 83.

** These changes from License Amendments 89 & 88.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

01-44-A

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALI- BRATION</u>	<u>CHANNEL OPERA- TIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERA- TIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3, 4
3) Containment Pressure-High-High	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4 <u>01-23-A</u>
c. Containment Ventilation Isolation								<u>02-20-A</u>
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3, 4
2) Deleted								
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4) Containment Ventilation Exhaust Radiation-High (RM-44A and 44B)	S	R(6)	Q(2)	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

01-23-A

02-35-A

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALI-BRATION	CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	<u>01-44-A</u>
4. Steam Line Isolation									
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3	
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3	
3 c. Containment Pressure-High-High	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	1, 2,	<u>01-23-A</u>
d. Steam Line Pressure-Low	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3	<u>01-23-A</u>
e. Negative Steam Line Pressure Rate-High	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	3(3)	<u>01-23-A</u>
5. Turbine Trip and Feedwater Isolation									
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2	
b. Steam Generator Water Level-High-High	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	1, 2	<u>01-23-A</u>
6. Auxiliary Feedwater									
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3	
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	1, 2, 3	
c. Steam Generator Water Level-Low-Low									
1) Steam Generator Water Level-Low-Low	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3(5)	<u>01-23-A</u>
2) RCS Loop ΔT	N.A.	R(6)	Q	N.A.	N.A.	N.A.	N.A.	1, 2	<u>01-23-A</u>

DIABLO CANYON - UNITS 1 & 2

3/4 3-34

Unit 1 Amendment No. 103
 Unit 2 Amendment No. 102
 July 2, 1995

TAB10.4A

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALI-BRATION	CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERA-TIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	WHICH SURVEILLANCE IS REQUIRED	MODES FOR	01-44-A
6. Auxiliary Feedwater (Continued)										
d. Undervoltage - RCP	N.A.	R(6)	N.A.	R	N.A.	N.A.	N.A.	±		01-23-A
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.									
7. Loss of Power										
a. 4.16 kV Emergency Bus Level 1	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4		02-11-A
b. 4.16 kV Emergency Bus Level 2	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4		
8. Engineered Safety Feature Actuation System Interlocks										
a. Pressurizer Pressure, P-11	N.A.	R(6)	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3		01-23-A
b. DELETED										
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	(6) N.A.	N.A.	N.A.	1, 2, 3		01-23-A
9. Residual Heat Removal pump trip - low RWST level	S	R(6)	N.A.	R	N.A.	N.A.	N.A.	N.A.		01-29-M 01-23-A

TABLE NOTATIONS

- (1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
- (2) For the Containment Ventilation Exhaust Radiation - High monitor only, a CHANNEL FUNCTIONAL TEST shall be performed at least once every 3192 days.
- (3) Trip function automatically blocked above P-11 (Pressurizer Pressure Interlock) setpoint and is automatically blocked below P-11 when Safety Injection on Steam Line Pressure-Low is not blocked.
- (4) DELETED
- (5) For Mode 3, the Trip Time Delay associated with the Steam Generator Water Level-Low-Low channel must be less than or equal to

464.1 seconds.

(6) Includes verification of time constants where applicable.

(new)

01-23-A

INSTRUMENTATION

03-01-A

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION: ¶

01-01-A

- a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, ~~adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.~~
- b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.
- c. The provisions of Specification 3.0.3 are not applicable ~~for the Fuel Building Radioactivity instrumentation.~~

03-02-M

03-06-A

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

~~(New) * Separate condition entry is allowed for each function.~~

01-01-A

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

INSTRUMENT	MINIMUM REQUIRED CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION	
1. Fuel Handling Building					01-43-A
(New) Manual	2	**	NA	32**	03-01-A
a. Storage Area					03-08-M
1) Spent Fuel Pool	1	***	≤ 75 mR/hr	30 & 32**(a)	03-03-LG
2) New Fuel Storage	1	***	≤ 15 mR/hr	30 & 32**(a)	03-07-LS16
b. Gaseous Activity					
Fuel Handling Building	42	***	Per the ODCP	32**	03-07-LS16
Ventilation Mode Change(b)					
2. Control Room					03-01-A
Ventilation Mode Change	2***	All, and during		34, 36	03-04-M
a. Manual Initiation	2	movement of irradiated			03-08-M
b. Automatic Actuation Logic and Actuation Relays	2	fuel assemblies			03-15-M
c. Control Room Atmosphere Intake Radiation Monitors	2***		≤ 2 mR/hr		
3. Containment					03-01-A
a. Gaseous Activity					
1) Deleted					
2) RCS Leakage	1	1, 2, 3, 4	N.A.	31	03-01-A
3) Containment Ventilation Isolation (RM-44A or 44B)	1	6	Per the ODCP	33, 37	03-14-LS29
b. Particulate Activity					
1) Containment Ventilation Isolation (RM-44A or 44B)	1	6	Per the ODCP	33, 37	03-14-LS29
2) RCS Leakage	1	1, 2, 3, 4	N.A.	31	03-01-A
					03-03-LG
					03-07-LS16
					03-10-LG
					03-01-A
					03-01-A

*With fuel in the spent fuel pool or new fuel storage vault.

**With irradiated fuel in the spent fuel pool. During movement of irradiated fuel assemblies in the fuel handling building.

***One channel for each normal intake to the Control Room Ventilation System (common to both units).

(a) Action 32 is not applicable to the Fuel Storage Area Monitors following installation of RM-45A and 45B.

(b) The requirements for Fuel Handling Building Ventilation Mode Change are applicable following installation of RM-45A and 45B.

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 30 -	With less than the Minimum Required Channels OPERABLE requirement, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint or an individual qualified in radiation protection procedures with a radiation dose rate monitoring device is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel storage pool areas.	<u>01-43-A</u>
ACTION 31 -	With the number of OPERABLE channels less than required by the Minimum Required Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1. The provisions of Specification 3.0.4 are not applicable.	<u>03-01-A</u> <u>01-43-A</u>
ACTION 32 -	With the number of OPERABLE channels less than required by the Minimum Required Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.	<u>01-43-A</u>
ACTION 33 -	With the number of OPERABLE channels less than required by the Minimum Required Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.	<u>01-43-A</u>
ACTION 34 -	With the number of OPERABLE channels one less than required by the Minimum Required Channels OPERABLE requirement, within 4-hour 7 days initiate and maintain operation of the Control Room Ventilation System in a recirculation the pressurization mode with the HEPA filter and charcoal adsorber bank in operation; or when in MODE 1-4 be in MODE 3 in 6 hours and in MODE 5 in 36 hours; or during fuel movement, immediately suspend CORE ALTERATIONS and movement of irradiated fuel assemblies; or when in MODE 5 or 6, immediately initiate actions to restore one CRVS train to OPERABLE status.	<u>01-43-A</u> <u>03-05-LS14</u> <u>03-15-M</u>
(New) ACTION 35	With no OPERABLE channels, immediately place one CRVS train in the pressurization mode and enter the applicable ACTIONS for one train made inoperable by CRVS actuation instrumentation; or when in MODE 1-4 be in MODE 3 in 6 hours and in MODE 5 in 36 hours; or during fuel movement, immediately suspend CORE ALTERATIONS and movement of irradiated fuel assemblies; or when in MODE 5 or 6, immediately initiate actions to restore one CRVS train to OPERABLE status.	<u>03-15-M</u>
(New) ACTION 37	With the number of OPERABLE channels one less than required by the Required Channels, restore the affected channel to OPERABLE status within 4 hours.	<u>03-14-LS29</u>

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

	CHANNEL CHECK	CHANNEL CALIBRATION	ACTUATION LOGIC TEST	A D O I	CHANNEL FUNCTIONAL TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
1. Fuel Handling Building									01-44-A
(New) Manual	NA	NA	NA	R ^(a)	NA	NA	NA		03-01-A
a. Storage Area									03-08-M
1) Spent Fuel Pool	S	R	NA	NA	Q	NA	NA	:	
2) New Fuel Storage	S	R	NA	NA	Q	NA	NA	:	
b. Gaseous Activity									
Fuel Handling Building	S	R	NA	NA	Q	NA	NA	:	
Ventilation Mode Change(a)									
2. Control Room									03-01-A
Ventilation Mode Change	S	R			Q			All	
a. Manual Initiation	NA	NA	NA	R ^(a)	NA	NA	NA		03-08-M
b. Automatic Actuation Logic and Actuation Relays	NA	NA	M ^(a)	A	NA	M ^(a)	R		
c. Control Room Atmosphere									
Air Intake Radiation	S	R	NA	NA	Q	NA	NA		
3. Containment									
a. Gaseous Activity									
1) Deleted									
2) RCS Leakage	S	R	NA	NA	Q	NA	NA	1,2,3,4	03-01-A
3) Containment Ventilation Isolation (RM-44A or 44B)	S	R	NA	NA	Q	NA	NA	6	
b. Particulate Activity									
1) Containment Ventilation Isolation (RM-44A or 44B)	S	R	NA	NA	Q	NA	NA	6	
2) RCS Leakage	S	R	NA	NA	Q	NA	NA	1,2,3,4	03-01-A

^a With fuel in the spent fuel pool or new fuel storage vault.

(a) The requirements for Fuel Handling Building Ventilation Mode Change are applicable following installation of RM-45A and 45B.

(New) (b) Each train shall be tested at least once every 62 days on a STAGGERED TEST BASIS.

(New) (c) Verification of setpoint is not required.

INSTRUMENTATION

MOVABLE INCORE DETECTORS

04-01-R

LIMITING CONDITION FOR OPERATION

~~3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:~~

- ~~a. At least 75% of the detector thimbles,~~
- ~~b. A minimum of two detector thimbles per core quadrant, and~~
- ~~c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.~~

~~APPLICABILITY: When the Movable Incore Detection System is used for:~~

- ~~a. Recalibration of the Excore Neutron Flux Detection System, or~~
- ~~b. Monitoring the QUADRANT POWER TILT RATIO, or~~
- ~~c. Measurement of F_{NH} , $F_Q(Z)$ and F_{xy} .~~

ACTION:

~~With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

~~4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:~~

- ~~a. Recalibration of the Excore Neutron Flux Detection System, or~~
- ~~b. Monitoring the QUADRANT POWER TILT RATIO, or~~
- ~~c. Measurement of F_{NH} , $F_Q(Z)$ and F_{xy} .~~

INSTRUMENTATION

SEISMIC INSTRUMENTATION

05-01-R

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation# shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

- a. With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.
- b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION, as applicable, performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

#The seismic monitoring instrumentation is common to both units but located in Unit 1 or common areas.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

<u>INSTRUMENTS AND SENSOR LOCATIONS</u>	<u>MEASUREMENT RANGE</u>	<u>MINIMUM INSTRUMENTS OPERABLE</u>
1. <u>Triaxial Time-History Accelerographs</u>		
a. <u>Containment Base Slab, EI 89, 180°</u>	<u>± 1g</u>	<u>1*</u>
b. <u>Top Unit 1 Containment, EI 303.5, 225°</u>	<u>± 2g</u>	<u>1</u>
c. <u>Aux Building, EI 64</u>	<u>± 1g</u>	<u>1</u>
2. <u>Triaxial Peak Accelerographs</u>		
a. <u>Containment Base Slab, EI 89, 180°</u>	<u>± 2g</u>	<u>1</u>
b. <u>Top Unit 1 Containment, EI 303.5, 225°</u>	<u>± 5g</u>	<u>1</u>
c. <u>Intake near ASW Pump 1-2 Bay, EI 2</u>	<u>± 2g</u>	<u>1</u>
d. <u>Turbine Building, EI 85, Machine Shop</u>	<u>± 2g</u>	<u>1</u>
e. <u>Aux Building, EI 140, Hot Shop</u>	<u>± 2g</u>	<u>1</u>
f. <u>Aux Building, EI 140, Near Control Room Door</u>	<u>± 2g</u>	<u>1</u>
3. <u>Triaxial Response Spectrum Recorders</u>		
<u>Containment Base Slab, EI 89, 180°</u>	<u>1.6 - 90 g</u> <u>2-25.4 Hz</u>	<u>1</u>

*With reactor control room indications or annunciation.

TABLE 4.3.4

05-01-R

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENTS AND SENSOR LOCATIONS</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>CHANNEL FUNCTIONAL TEST</u>
1. Triaxial Time History Accelerographs			
a. Containment Base Slab, EI 89, 180**	M*	R	SA
b. Top Unit 1 Containment, EI 303.5, 225°	M*	R	SA
c. Aux Building, EI 64	M*	R	SA
2. Triaxial Peak Accelerographs			
a. Containment Base Slab, EI 89, 180°	N.A.	R***	N.A.
b. Top Unit 1 Containment, EI 303.5, 225°	N.A.	R***	N.A.
c. Intake near ASW Pump 1-2 Bay, EI 2	N.A.	R***	N.A.
d. Turbine Building, EI 85, Machine Shop	N.A.	R***	N.A.
e. Aux Building, EI 140, Hot Shop	N.A.	R***	N.A.
f. Aux Building, EI 140, Near Control Room Door	N.A.	R***	N.A.
3. Triaxial Response Spectrum Recorders			
Containment Base Slab, EI 89, 180°	N.A.	R***	N.A.

*Except seismic trigger.

**With reactor control room indications or annunciation.

***Channel calibration shall be in accordance with ANSI/ANS-2.2-1978.

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

06-01-R

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels# shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

- a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.
- b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-5.

#The meteorological monitoring instrumentation channels are common to both units.

TABLE 3.3-8

06-01-R

METEOROLOGICAL MONITORING INSTRUMENTATION

METEOROLOGICAL TOWER MINIMUM
INSTRUMENT AND LOCATION OPERABLE

1. Wind Speed

a. Nominal Elev. 25' 1 of 2
b. Nominal Elev. 250'

2. Wind Direction

a. Nominal Elev. 25' 1 of 2
b. Nominal Elev. 250'

3. Air Temperature - ΔT

a. Nominal Elev. 250' - 25' 1 of 2
b. Nominal Elev. 150' - 25'

TABLE 4.3-5

06-01-R

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL</u> <u>CHECK</u>	<u>CHANNEL</u> <u>CALIBRATION</u>
1. <u>Wind Speed</u>		
a. <u>Nominal Elev. 25'</u>	<u>D</u>	<u>SA</u>
b. <u>Nominal Elev. 250'</u>	<u>D</u>	<u>SA</u>
2. <u>Wind Direction</u>		
a. <u>Nominal Elev. 25'</u>	<u>D</u>	<u>SA</u>
b. <u>Nominal Elev. 250'</u>	<u>D</u>	<u>SA</u>
3. <u>Air Temperature - ΔT</u>		
a. <u>Nominal Elev. 250'-25'</u>	<u>D</u>	<u>SA</u>
b. <u>Nominal Elev. 150'-25'</u>	<u>D</u>	<u>SA</u>

INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation and control functions shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

- a. With less than the minimum required Function(s) of Table 3.3-9 operable, restore the inoperable Function(s) to OPERABLE status within 30 days or be in ~~MODE 3~~ within 6 hours and ~~HOT SHUTDOWN~~ within the next 12 hours.
- b. The provisions of Specification 3.0.4 are not applicable.
- c. Separate entry into Action a. is allowed for each Function in Table 3.3-9.

07-05-A

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-6.

4.3.3.5.2 Verify each required control circuit and control transfer switch is capable of performing the intended function at least once every 18 months.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
AND CONTROLS

<u>INSTRUMENT/CONTROL FUNCTION</u>	<u>READOUT/CONTROL LOCATION</u>	<u>REQUIRED NUMBER OF CHANNELS</u>	<u>07-06-LG</u>
1. Reactor Trip Breaker Indication	Reactor Trip Breaker	1/trip breaker	
2. Pressurizer Pressure	Hot Shutdown Panel	1	
3. Pressurizer Level	Hot Shutdown Panel	1	
4. Steam Generator Pressure	Hot Shutdown Panel	1/stm. gen.	
5. Steam Generator Wide Range Water Level or Auxiliary Feedwater Flow	Hot Shutdown Panel	1/stm. gen.	<u>07-10-LS26</u>
6. Condensate Storage Tank Water Level	Hot Shutdown Panel	1	
7. Auxiliary Feedwater Flow	Hot Shutdown Panel	4/stm. gen.	<u>07-10-LS26</u>
8. Charging Flow	Hot Shutdown Panel	1	
9. RCS Loop 1 Temperature Indication	Dedicated Shutdown Panel	Hot and Cold Leg Temperature Indication	
10. Auxiliary Feedwater Flow Control - AFW Pump and Associated Valves - Transfer Switches	Hot Shutdown Panel 4kV Switchgear	any 2 of 3 AFW pumps	<u>07-06-LG</u>
11. Charging Flow Control - Centrifugal Charging Pump - Transfer Switch	Hot Shutdown Panel 4kV Switchgear	2 of 2 pumps	<u>07-06-LG</u>
12. Component Cooling Water Control - Component Cooling Water Pump - Transfer Switch	Hot Shutdown Panel 4kV Switchgear	any 2 of 3 CCW pumps	<u>07-06-LG</u>
13. Auxiliary Saltwater Control - Auxiliary Saltwater Pump - Transfer Switch	Hot Shutdown Panel 4kV Switchgear	2 of 2 pumps	<u>07-06-LG</u>
14. Emergency Diesel Generator Control - EDG Start	EDG Local Control Panel	3 of 3 EDGs	<u>07-06-LG</u>

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>
1. Reactor Trip Breaker Indication	N.A.	N.A.
2. Pressurizer Pressure	M	R
3. Pressurizer Level	M	R
4. Steam Generator Wide Range Water Level	M	R
5. Steam Generator Pressure	M	R
6. Condensate Storage Tank Water Level	M	R
7. Auxiliary Feedwater Flow	M	R
8. Charging Flow	M	R
9. RCS Loop 1 Temperature Indication	M	R

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels functions shown in Table 3.3-10 shall be OPERABLE.

08-01-A

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

- a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels but at least one accident monitoring channel OPERABLE shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours, prepare and submit a Special Report to the alternate method of monitoring the appropriate parameter(s) cause of the inoperability, and plans and schedule for restoring the channel to OPERABLE status. 08-11-LS30
- b. With the number of OPERABLE accident monitoring instrumentation channels for one or more instrument functions except the containment recirculation sump level narrow range, the main steam line radiation monitor, the containment area radiation monitor high range, and the plant vent radiation monitor high range less than the Minimum Channels OPERABLE requirements of Table 3.3-10, except for the Containment Hydrogen Concentration, restore at least one the inoperable channel(s) to OPERABLE status within 48 hours 7 days or be in at least HOT SHUTDOWN within the next 12 hours, enter the Action Required references in Table 3.3-10. 08-11-LS30
- c. With the number of OPERABLE channels for the containment recirculation sump level narrow range less than the Minimum Channels OPERABLE requirement As required by the Action Requirements of Table 3.3-10, except for the Containment Hydrogen Concentration monitors, restore the inoperable channel to OPERABLE status within 30 days or be in at least HOT STANDBY within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours. 08-11-LS30
- d. With the number of OPERABLE channels for the main steam line radiation monitor, or the containment area radiation monitor high range or the plant vent radiation monitor high range less than the Minimum Channels OPERABLE requirements As required by the Action Requirements of Table 3.3-10, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days that provides actions taken, identifies the alternate method of monitoring the appropriate parameter(s) cause of the inoperability and plans and schedule for restoring the channels to OPERABLE status. 08-11-LS30
08-04-LS17
- e. The provisions of Specification 3.0.4 are not applicable.

(New) Separate Condition entry is allowed for each function

01-01-A

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK ~~once per 31 days for each required instrument that is normally energized~~ and CHANNEL CALIBRATION ~~at the frequencies shown in Table 4.3-7, once per 18 months;~~

08-11-LS30

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT FUNCTION	REQUIRED NO. OF CHANNELS	ACTION REQUIREMENTS		MINIMUM CHANNELS OPERABLE	08-01-A 08-03-A 08-11-LS30
		FROM ACTION b			
1. Containment Pressure (normal range)	2	E	4		
(New) Containment Pressure (wide range)	2	C			08-11-LS30
2. Reactor Coolant Outlet Temperature - T _{hot} (Wide Range)	2 (1/loop in two loops)	C	4/loop in one loop		
3. Reactor Coolant Inlet Temperature - T _{cold} (Wide Range)	2 (1/loop in two loops)	C	4/loop in one loop		
4. Reactor Coolant Pressure - Wide Range	2	E	4		
5. Pressurizer Water Level	2	C	4		
6. Steam Line Pressure	2/steam generator	C	4/steam generator		
7. Steam Generator Water Level - Narrow Range	2/steam generator	C	4/steam generator		
(NEW) Steam Generator Water Level - Wide Range	1/steam generator	C			08-11-LS30
8. Refueling Water Storage Tank Water Level	2	E	4		
9. Containment Reactor Cavity Sump Level-Wide Range	2	E	4		
10. Containment Recirculation Sump Level-Narrow Range	N.A. 2	C	4		
11. Auxiliary Feedwater Flow Rate	1/steam generator	E	4/steam generator		
12. Reactor Coolant System Subcooling Margin Monitor	4		4		08-11-LS30
13. PORV Position Indicator	2*valve		4valve**		08-11-LS30
14. PORV Block Valve Position Indicator	4valve		4valve		08-11-LS30
15. Safety Valve Position Indicator	2***valve		4valve		08-11-LS30
16. In Core Thermocouples Quadrant 1	4/core quadrant	E	2/core-quadrant		08-11-LS30
In Core Thermocouples Quadrant 2	4/core quadrant	C			
In Core Thermocouples Quadrant 3	4/core quadrant	C			
In Core Thermocouples Quadrant 4	4/core quadrant	C			
17. Main Steam Line Radiation Monitor	N.A.		4/steam-line		
18. Containment Area Radiation Monitor-High Range	N.A. 2	D	4		08-11-LS30
19. Plant Vent Radiation Monitor-High Range	N.A.		4		08-11-LS30

- ~~*One direct, stem mounted indicator per valve and one temperature element in the common discharge line from the PORVs.~~
- ~~**One common temperature element is equivalent to 1/valve for all PORVs.~~
- ~~***One acoustic monitor and one temperature element.~~

(new) Neutron Flux/Wide Range NIS	2	C
(new) Containment Isolation Valve Position	1/valve ^{(a)(b)}	C
(new) Containment Hydrogen Concentration	2	C
(new) Condensate Storage Tank Level	2	C

08-11-LS30
 08-11-LS30
 08-11-LS30
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- (new) (a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated valve, closed manual valve, blind flange, or check valve with flow through the valve secured.
- (new) (b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.
- (new) (c) Neutron detectors are excluded from channel calibration.
- (new) (d) A channel consists of two Incore Thermocouples.

08-11-LS30
 08-11-LS30
 08-11-LS30
 08-11-LS30

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature T_{hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature T_{cold} (Wide Range)	M	R
4. Reactor Coolant Pressure Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level Narrow Range	M	R
8. Refueling Water Storage Tank Water Level	M	R
9. Containment Reactor Cavity Sump Level Wide Range	M	R
10. Containment Recirculation Sump Level Narrow Range	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Reactor Coolant System Subcooling Margin Monitor	M	R
13. PORV Position Indicator	M	R
14. PORV Block Valve Position Indicator	M	R
15. Safety Valve Position Indicator	M	R
16. In Core Thermocouples	M	R
17. Main Steam Line Radiation Monitor	M	R
18. Containment Area Radiation Monitor High Range	M	R*
19. Plant Vent Radiation Monitor High Range	M	R
20. Reactor Vessel Level Indication System	M	R

*CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

08-11-LS30

INSTRUMENTATION

CHLORINE DETECTION SYSTEMS

11-01-R

LIMITING CONDITION FOR OPERATION

~~3.3.3.7 Two independent Chlorine Detection Systems, # with their Alarm/Trip Setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE.~~

~~APPLICABILITY: All MODES, when bulk chlorine gas is stored on the plant site.~~

ACTION:

- ~~a. With one Chlorine Detection System inoperable, restore the inoperable system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room Ventilation System in a recirculation mode with the HEPA filter and charcoal absorber system in operation.~~
- ~~b. With both Chlorine Detection Systems inoperable, within 1 hour initiate and maintain operation of the Control Room Ventilation System in a recirculation mode with the HEPA filter and charcoal absorber system in operation.~~

SURVEILLANCE REQUIREMENTS

~~4.3.3.7 Each Chlorine Detection System shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours and a CHANNEL FUNCTIONAL TEST at least once per 31 days. At least once per 18 months, the following inspections and maintenance shall be performed:~~

- ~~a. Check constant head bottle level and refill as necessary.~~
- ~~b. Clean the sensing cells.~~
- ~~c. Check flow meter operation and clean or replace filters and air lines as necessary.~~
- ~~d. Check air pump for proper operation, and~~
- ~~e. Verify that the detector responds to chlorine.~~

~~#The Chlorine Detection System is common to both units and is installed in the normal intakes to the Control Room Ventilation System.~~

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INSTRUMENTATION

EXPLOSIVE GAS EFFLUENT MONITORING INSTRUMENTATION

09-01-LG

LIMITING CONDITION FOR OPERATION

3.3.3.9 Deleted

3.3.3.10 The explosive gas monitoring instrumentation channels (ANR-75 or ANR-76) shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.2.5 are not exceeded.

APPLICABILITY: During GASEOUS RADWASTE SYSTEM operation.

ACTION:

- a. With an explosive gas monitoring instrumentation channel (ANR-75 or ANR-76) Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable and follow action b. below.
- b. With only one OPERABLE explosive gas monitoring instrumentation channel, operation of this system may continue for up to 14 days. After 14 days or with no channels OPERABLE, operation of this system may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.
- c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Deleted

4.3.3.10.1 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of a daily CHANNEL CHECK and a monthly CHANNEL FUNCTIONAL TEST.

4.3.3.10.2 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of a quarterly CHANNEL CALIBRATION. The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

- a. Two volume percent oxygen, balance nitrogen, and
- b. Four volume percent oxygen, balance nitrogen.

INSTRUMENTATION

3/4.3.4 TURBINE OVERSPEED PROTECTION

10-01-R

LIMITING CONDITION FOR OPERATION

~~3.3.4.1 At least one Turbine Overspeed Protection System shall be OPERABLE.~~

~~APPLICABILITY: MODES 1, 2 and 3 (during turbine operation).~~

ACTION:

- ~~a. With one stop valve or one control valve per high pressure turbine steam line inoperable or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or isolate the turbine from the steam supply within the next 6 hours.~~
- ~~b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.~~

SURVEILLANCE REQUIREMENTS

~~4.3.4.1.1 The provisions of Specification 4.0.4 are not applicable.~~

~~4.3.4.1.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:~~

- ~~a. At least once per quarter by cycling and direct observation of the movement of each of the following valves through at least one complete cycle from the running position:
 - ~~1) Four high pressure turbine stop valves,~~
 - ~~2) Four high pressure turbine control valves,~~
 - ~~3) Six low pressure turbine reheat stop valves, and~~
 - ~~4) Six low pressure turbine reheat intercept valves.~~~~
- ~~b. At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems.~~
- ~~c. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.~~

Methodology For Mark-Up of Current TS

This enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. Deletions - Material is no longer in the specifications. (this includes material which is moved to the Bases of the TS).
2. Additions - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. Modifications - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. Administrative - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is :

- | | |
|------------------|---|
| Deletions - | The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin. |
| Additions - | The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin. |
| Modifications - | The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin. |
| Administrative - | The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the markup of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS. |

Methodology For Mark-Up of Current TS

(Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS markup. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers

(1 Page)

Description of Changes

(22 Pages)

TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS

SECTION 3.3

Technical Specification Title	CHG NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Reactor Trip System Instrumentation	1	3.3.1	3.3.1	3.3.1	3.3.1
ESFAS Instrumentation	2	3.3.2	3.3.2	3.3.2	3.3.2
Radiation Monitoring for Plant Operations	3	3.3.3.1	3.3.3.1	3.3.3.1	3.3.3.1
Movable Incore Detectors	4	NA	NA	NA	3.3.3.2
Seismic Instrumentation	5	NA	NA	NA	3.3.3.3
Meteorological Instrumentation	6	NA	NA	NA	3.3.3.4
Remote Shutdown Instrumentation	7	3.3.3.5	3.3.3.5	3.3.3.2	3.3.3.5
Accident Monitoring Instrumentation	8	3.3.3.6	3.3.3.6	3.3.3.3	3.3.3.6
Explosive Gas Monitoring Instrumentation	9	NA	NA	3.3.3.4	3.3.3.9
Turbine Overspeed Protection	10	NA	NA	3.3.4	3.3.4.1
Chlorine Detection Systems	11	NA	NA	NA	3.3.3.7

DESCRIPTION OF CHANGES TO TS SECTION 3/4.3

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	A	A Note, "Separate Condition entry is allowed for each Function," is added to the ACTIONS for the Reactor Trip System, ESFAS, [Remote Shutdown also applies to each required ASP control], and Accident Monitoring Instrumentation. This change clarifies those situations where the current TS ACTION Statements are not uniquely associated with a particular Function or where the required channels are specified on a per steam line, per loop, per SG, per bus, etc., basis. This change is consistent with current operating practices and NUREG-1431. []
01-02	LG	The CTS require that response time testing be performed on each reactor trip and ESFAS function every 18 months and that alternate trains be tested in successive tests. The CTS description of the channel testing protocol matches the improved TS definition of STAGGERED TEST BASIS. However, several trip functions do not require response time testing, as indicated by N.A. in the tables of response time limits [(presently located in Tables 3.3-2 and 3.3-5 of the CTS, which are being to the FSAR per CN 01-35-LG)]. The improved TS specify that required response time testing be performed on a STAGGERED TEST BASIS and do not impose any requirements as to which train should be tested. Therefore, the word "requirement" is added to the CTS and the requirement to ensure that each train is tested every 36 months is moved to the Bases for ITS SR 3.3.1.16 and SR 3.3.2.10.
01-03	LS1	In CTS SR 4.3.1.2 and 4.3.2.2, the active verb is changed from "demonstrated" to "verified." This allows Reactor Trip System and ESFAS sensor response time verifications to be performed per WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements." This change is consistent with Traveler TSTF-111 Rev. 1, which revises the Bases for ITS SR 3.3.1.16 and SR 3.3.2.10 to allow the elimination of pressure sensor response time testing.

CHANGE
NUMBER

NSHC

DESCRIPTION

01-04

LG

In CTS Tables [3.3-1 and 3.3-3], the ["Channels to Trip" and] "Minimum Channels OPERABLE" columns are deleted, consistent with NUREG-1431. [The channel trip logic is moved to the Bases.] ACTION Statements are also revised to delete references to "Minimum Channels OPERABLE" requirements. The ITS terminology, "Required Channels," is used. The logic coincidence information has been moved to the ITS Bases for each applicable specification. OPERABILITY requirements for each Function are addressed by the Required Channels and applicable Conditions in the ITS.

01-05

A

The LCO 3.0.4 exception [footnote #] in CTS Table 3.3-1 is deleted entirely. ACTION Statement [8] in CTS Table 3.3-1 permits continued operation for an unlimited period of time. Therefore, no exception to ITS LCO 3.0.4 is needed for this ACTION Statement. [] This change is consistent with NUREG 1431.

01-06

LS2

Consistent with NUREG-1431, a new ACTION Statement [2.1] is created which is essentially the same as current ACTION Statement [6], and is similar to current ACTION Statement [2], but does not require a reduction in THERMAL POWER to less than 75% RTP or the measurement of the QPTR if above 75% RTP. This new ACTION Statement is applied to the Power Range Neutron Flux, High Positive Rate [, High Negative Rate] trip and Low Setpoint trip functions. The latter is an administrative change only since its Trip Setpoint of 25% RTP and APPLICABILITY (below P-10) are well below the reduced power of 75% RTP. The Power Range Neutron Flux, High Positive Rate [and High Negative Rate] trip[s] are rate functions and their effectiveness is not improved by reducing power; therefore, this new ACTION Statement is also applied to [these trips], as discussed in LS-2.

CHANGE
NUMBER

NSHC

DESCRIPTION

01-07

LS3

With one intermediate range neutron flux channel inoperable, current ACTION Statement [3.a] applies below the P-6 interlock. For those times that the plant is above P-6 but below 10% RTP (the P-10 interlock setpoint), current ACTION Statement [3.b] applies. ACTION Statement [3.b] is revised to establish a 24 hour Completion Time for channel restoration or changing the power level to either below P-6 or above P-10. The intermediate range neutron flux channels provide protection between these power levels and the APPLICABLE MODES have been revised to indicate this via new footnote [(d)]. With the revised APPLICABILITY, current ACTION Statement [3.a] is deleted since it is outside the new APPLICABILITY. The source range neutron flux detectors provide protection below P-6 and the power range neutron flux detectors provide protection above P-10. The addition of the 24 hour Completion Time (CTS has no Completion Time) limits the window of operation during which the intermediate range neutron flux trip function provides protection in a 1 of 1 logic configuration and ensures the low probability of occurrence of a reactivity transient during this time period that would require an intermediate range flux trip. Although this change is less restrictive since a power increase is an allowed option, the ACTION Statement would ensure protection by entering the range of the four power range neutron flux channels.

With both intermediate range neutron flux channels inoperable in MODE 1 (below P-10) and MODE 2 (above P-6), LCO 3.0.3 would be entered under the CTS and the plant would have to be in MODE 3 within 7 hours. With both intermediate channels inoperable, new ACTION Statement [3.1] requires immediate suspension of operations involving positive reactivity additions and a power reduction below P-6 within 2 hours. New ACTION Statement [3.1] is less restrictive since a reduction to MODE 3 would no longer be required; however, the CTS are overly conservative in this area. Below P-6 the source range channels provide protection; therefore, the required ACTION for both intermediate range channels inoperable should be to exit plant conditions where this trip function provides protection. New ACTION Statement [3.1] will preclude any power level increase and require a controlled power reduction to less than P-6 where the source range channels provide protection. The 2 hour Completion Time ensures the low probability of occurrence of an event during this period that may require the protection afforded by the intermediate range neutron flux trip. These actions actually provide a more timely and appropriate redress to the condition than entering LCO 3.0.3.

These changes are consistent with NUREG-1431.

CHANGE
NUMBER

NSHC

DESCRIPTION

01-08

M

Current ACTION Statement [4], for one source range channel inoperable in MODE 2 below P-6, is revised to require the immediate suspension of operations involving positive reactivity changes. The CTS does not specify the timing for this ACTION. A new ACTION Statement [4.1] is added to address the condition where both source range channels are inoperable in MODE 2 below P-6 []. The new ACTION Statement [4.1] requires that the RTBs be opened immediately. This action essentially accomplishes the reactor trip function. In such a condition with the CTS, LCO 3.0.3 would have been entered. This is more conservative since immediate rod insertion will take the plant to MODE 3 [] much more quickly than[the 7 hours] allowed by LCO 3.0.3. These changes are consistent with NUREG-1431.

01-09

M

This change revises CTS ACTION 5 to require the immediate suspension of positive reactivity additions. Reference to specification 3.1.1.2 is deleted since the ITS reference is only to 3.1.1.1 which is more conservative for MODE 5. These changes are consistent with NUREG-1431.

01-10

LS32

Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).

01-11

LS5

Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

01-12

M

New ACTION Statement [8.1] is created to differentiate between those RTS interlocks required to be operable in MODE 1 only, and those interlocks required to be operable in MODES 1 and 2. If the interlock function is required to be operable in MODE 1 only and the LCO and ACTION requirements are not met, then new ACTION Statement [8.1] requires that the unit be taken to MODE 2 within 7 hours.

In addition, current ACTION Statement [8] is revised for those interlocks required to be OPERABLE in MODES 1 and 2. If one channel is inoperable, the interlock must be determined to be in its required state or the plant must be in at least HOT STANDBY within 7 hours.

The changes to current ACTION Statement [8] and the addition of new ACTION Statement [8.1] are more restrictive, consistent with NUREG-1431. Current ACTION Statement [8] will continue to apply to Functional Units [22.a and 22.e.] Revised ACTION Statement [8] and new ACTION Statement [8.1] provide one less hour to exit APPLICABILITY, i.e. 7 hours, than the current ACTION Statement [8] which has the 1 hour interlock state verification or entry into LCO 3.0.3 which allows an additional 1 hour plus 6 hours to exit APPLICABILITY, for a total of 8 hours.

CHANGE
NUMBER

NSHC

DESCRIPTION

01-13

LS6

[ACTION Statement [10] is revised to note that the 2 hour [train and] reactor trip breaker bypass allowance for [train or] breaker surveillance testing can also be used for maintenance. This change does not impact the conclusions of WCAP-10271-P-A, Supplement 2, Rev. 1 since there is no change to the bypass time. This change is consistent with Traveler TSTF-168.] ACTION Statement [10] is [also] revised to require restoration of an inoperable RTB within 1 hour or the plant must be in HOT STANDBY within the next 6 hours, consistent with NUREG-1431. This is less restrictive since an additional hour is provided for the transition to MODE 3.

01-14

A

In the ISTS Table 3.3.1-1, Function 20, the Reactor Trip Breaker (RTB) Undervoltage and Shunt Trip Mechanisms are separate from the RTB Functional Unit. The CTS have been revised to reflect these requirements.

New [footnote (b) has] been added to the RTB Functional Unit to note that the same OPERABILITY requirements and ACTIONS apply to a bypass breaker if it is racked in and closed for bypassing an RTB. The bypass breakers were already handled in this fashion. ACTION [12] in CTS Table 3.3-1 has been revised accordingly.

01-15

A

Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).

01-16

LS40

The requirement to verify the setpoint during the quarterly TADOT for RCP Underfrequency [and RCP Undervoltage] is deleted, consistent with NUREG-1431.

01-17

A

Consistent with NUREG-1431, LCO 3.3.1 Required ACTION D Note, CTS Table 3.3-1 ACTION Statement 2 and new ACTION Statement 2.1 have been modified by a Note that allows the bypass to be used for surveillance testing or setpoint adjustment. Setpoint adjustment can be performed at power and may be required by other Technical Specifications. The reason for placing the channel in bypass does not affect the impact of having the channel in bypass.

**CHANGE
NUMBER**

NSHC

DESCRIPTION

01-18

LS7

The CTS requirement to reduce the Power Range Neutron Flux Trip setpoint in the event a power range flux channel is inoperable is deleted. This deletion is consistent with NUREG-1431, and justified by:

- 1) The loss of one channel does not impact the reliability of the Reactor Trip System because the affected channel is placed in trip. It may, however, impact the tilt monitoring for a portion of the reactor core. If the plant wishes to remain at 100% RTP, then the QPTR must be measured using the movable incore detectors. Otherwise, the power level must be reduced to 75% RTP. If the plant chooses to reduce power rather than measure QPTR using the movable incore detectors, the peaking factor surveillances must still be performed on the required frequency.
- 2) The loss of one channel does not necessarily indicate any core tilt, but rather the inability to measure core tilt with the excore instrumentation. On this basis, there is no justification for reducing the Trip Setpoint, and incurring the potential for a reactor trip, when there is no indication of an abnormal condition existing in the core.

NUREG-1431 also allows 12 hours to reduce the thermal power to less than 75% RTP rather than the 4 hours required by the CTS.

If the Power Range Neutron Flux trip function is inoperable, but the input to the QPTR is operable, the ISTS do not require that the QPTR be monitored every 12 hours.

If the above ACTIONS are not completed, the plant must be in MODE 3 within 12 hours. (See also CN 1-53-A).

01-19

LS8

This change reflects a revision to current ACTION Statement [6]. If the requirements of current ACTION Statement [6] are not met, LCO 3.0.3 would be entered. In accordance with the ISTS, this ACTION Statement is revised to state that if the ACTION requirements are not met, the plant must be taken below the P-7 interlock setpoint within the next 6 hours. [The APPLICABLE MODES for Functional Units 9, 11, 12, 15, and 16 in CTS Table 3.3-1 are also revised to add new footnote (g).]

01-20

A

Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).

CHANGE
NUMBER

NSHC

DESCRIPTION

01-21

A

The monthly and quarterly channel calibrations associated with Notes (3), (4), and (6) of CTS Table 4.3-1 have been moved from the Power Range Neutron Flux-High Setpoint Function to the Overtemperature [ΔT] Function. This change clarifies the relationship of these surveillances to the f_1 (ΔI) penalty portion of the Overtemperature [ΔT] Function. The primary purpose of these surveillances is to verify correct f_1 (ΔI) input to Overtemperature [ΔT]. Although these surveillances affect all power range neutron flux channels, and appropriate action must be taken for any affected power range neutron flux channel, this change groups the surveillances with the most appropriate reactor trip function for OPERABILITY concerns.

[] The applicable portions of CTS Table 4.3-1 Notes (3) and (6) are incorporated directly into ITS SR 3.3.1.3 and SR 3.3.1.6, as discussed in CN 1-25-A. Note (4) has been deleted from the daily, monthly, and quarterly surveillances associated with Notes (2), (3), and (6) of CTS Table 4.3-1 since these surveillances are not CHANNEL CALIBRATIONS, rather they are comparisons and adjustments as needed. These changes are consistent with NUREG-1431.

01-22

M

Quarterly COTs have been added to CTS Table 4.3-1 for the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux trip functions in the event extended operation within their APPLICABILITY (i.e., MODE 1 below P-10 and MODE 2) takes place. The CTS only require a COT prior to startup for these functions. New Note [(19)] has been added to require that the new quarterly COT be performed within 12 hours after reducing power below P-10 for the power range and intermediate range instrumentation (P-10 is the dividing point marking the APPLICABILITY for these trip functions), if not performed within the previous 92 days. [In addition, new Note (20) has been added] such that the P-6 and P-10 interlocks are verified to be in their required state during all COTs on the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux trip functions. These changes are consistent with NUREG-1431 and traveler WOG-106.

01-23

A

This change adds new Note [(22)] to CTS Table 4.3-1 and new Note [(56)] to CTS Table 4.3-2 that explicitly require the 18-month calibrations to include verifications of affected time constants, consistent with NUREG-1431.

**CHANGE
NUMBER**

01-24

NSHC

LS9

DESCRIPTION

This change reflects a relaxation in the performance of COTs prior to startup, consistent with NUREG-1431. These COTs, for the Power Range Neutron Flux - Low, Intermediate Range Neutron Flux, and Source Range Neutron Flux trip functions, will not be required if performed within the previous 92 days. Note [(1)] of CTS Table 4.3-1 has been revised to extend this period from 31 days to 92 days. Note [(1a)] is added for use with the turbine trip function TADOTs, for which no such change was provided.

CHANGE
NUMBER

NSHC

DESCRIPTION

01-25

A

NUREG-1431 incorporates the CTS 4.0.4 exception for the surveillances covered by CTS Table 4.3-1 Notes (2), (3), and (6) into the ISTS SR 3.3.1.2, 3.3.1.3, and 3.3.1.6 surveillance frequencies. In the CTS, these surveillances have no time requirements [except for Note (2)]. In general, the combined effect of CTS 4.0.3 and CTS 4.0.4 would allow [24 hours] for the completion of a surveillance if it were not performed upon entry into the specified MODE or other condition for that surveillance. CTS 4.0.3 allows [24 hours] for the completion of a surveillance once the appropriate MODE or condition is established. In the improved TS, the TS 4.0.4 exceptions are incorporated as a NOTE in each affected surveillance. In converting to the NUREG-1431 format, ITS SR 3.3.1.2, SR 3.3.1.3, and SR 3.3.1.6 require these surveillances to be performed within [24] hours after THERMAL POWER is greater than or equal to 15% RTP (CTS Note (2) for the comparison of NIS power to calorimetric power), within [24] hours after 50% RTP (CTS Note (3) for the comparison of excore Axial Flux Difference (AFD) to incore AFD), and [after achieving equilibrium conditions (per CTS 4.2.2.2.d.1) with THERMAL POWER greater than or equal to 75%] RTP (CTS Note (6) for the incore-excore calibration), respectively. Since these surveillances are conditional on THERMAL POWER levels greater than the MODE 2 to MODE 1 breakpoint, there is no need to retain the CTS 4.0.4 exception, as discussed in ITS Section 1.4, as long as the ITS Surveillance Notes allow the plant to reach conditions appropriate for the performance of the surveillances.

The power level at which the monthly surveillance is performed to compare incore vs. excore AFD, and adjustment of the NIS as required by Note (3) of CTS Table 4.3-1, is changed from $\geq 15\%$ RTP to $\geq 50\%$ RTP. This change is also precipitated by the conversion to the ITS format which replaces the CTS 4.0.4 exception with a finite time interval after exceeding a specified power level. With the deletion of the CTS 4.0.4 exception, the specified power level in ITS SR 3.3.1.3 should reflect the applicable safety analysis basis consistent with the [APPLICABILITY and] Required ACTIONS of ITS LCO 3.2.3 (AFD) and LCO 3.2.4 (QPTR). As with the changes discussed above, there is no need to retain the CTS 4.0.4 exception as long as the ITS Surveillance Notes allow the plant to reach conditions appropriate for the performance of the surveillances.

These changes are considered administrative in nature in that they simply reflect the incorporation of the CTS 4.0.4 exceptions into the improved TS.

CHANGE
NUMBER

NSHC

DESCRIPTION

01-26	LG	This change moves the details concerning NIS detector calibration to the Bases for ITS SR 3.3.1.11, consistent with NUREG-1431. This information is more appropriately controlled outside of the TS while the calibration requirement itself and its Frequency are unchanged.
01-27	LS10	Surveillances on the Source Range Neutron Flux trip function are reorganized to reflect plant status in accordance with NUREG-1431. New Note [(19)] requires that the quarterly COT be performed within 4 hours after reducing power below the respective source range instrumentation Applicabilities, if not performed within the previous [92] days. Since the COT is valid for [92] days, there is no need to repeat it if one has been performed within the prior [quarter]. The 4 hour allowance permits a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the Reactor Trip Breakers are opened and this trip function no longer provides protection. Since the CTS has no Specification 4.0.4 exception, this 4 hour allowance is less restrictive.
01-28	A	Note [8] is revised to require the P-6 and P-10 interlock verification to be performed during all source range COTs. These permissives are verified to be in their correct state prior to entry into MODES 3, 4, and 5 during shutdown and after leaving MODES 3, 4, and 5 during startup. These changes are consistent with NUREG-1431.
01-29	LG	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-30	M	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-31	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-32	LG	[] [Note (10) of CTS Table 4.3-1 is deleted since it is redundant; every TADOT requires independent UVTA and STA verification per ITS SR 3.3.1.4, not just those TADOTs following maintenance or adjustment.] Notes [(14) and (16) applicable to the RTBs and the RTB bypass breakers] of CTS Table 4.3-1 are moved to the Bases for ITS SR 3.3.1.14. These changes are consistent with NUREG-1431.
01-33	TR1	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-34	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

CHANGE
NUMBER

NSHC

DESCRIPTION

01-35	LG	This change moves response time limit tables to the updated FSAR per Generic Letter 93-08 and NUREG-1431.
01-36	M	This change adds a requirement to perform a Channel Calibration on Functional Unit 17 every 18 months. This change is consistent with NUREG-1431.
01-37	A	In the CTS, the "Minimum Channels OPERABLE" is one less than the "Total Number of Channels" for Functional [22.c] (P-8), [22.d](P-9), and [22.e] (P-10) in Table 3.3-1 and Functional Unit [8.a] (P-11) in Table [3.3-3]. For these Reactor Trip System and ESFAS interlocks, current ACTION Statements [8 and 21] for an inoperable channel are based on the "Minimum Channels OPERABLE" columns in Tables 3.3-1 and [3.3-3]. In the improved TS, only the "Total Number of Channels" information is retained in the LCO and that column is relabeled as the "Required Channels", as discussed in CN 1-04-LG and CN 1-43-A. Required ACTIONS in improved TS 3.3.1 Conditions S and T and improved TS 3.3.2 Condition L are tied to the Required Channels. Therefore, the required permissive channels for these Functional Units are revised in the CTS. Refer also to CN 01-51-LG for P-7.
01-38	R	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-39	A	This change adds Note [5] to Functional Units [2.b,3, and 4] in CTS Table 4.3-1. Note [5] is currently listed against Functional Unit [2.a]. Testing methodology and the timing of that testing for the power range channels apply to all power range functions, not just power range-high. As such, this is an administrative change only. ITS SR 3.3.1.11 applies to all power range functions in a similar manner.
01-40	LS41	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-41	A	This change moves the first sentence of note (1) of Table 3.3-2 to ITS SR 3.3.1.16, and moves the rest of Note (1) to the Bases.
01-42	M	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

CHANGE
NUMBER

NSHC

DESCRIPTION

01-43	A	The "Total Number of Channels" columns in CTS Tables 3.3-1 and [3.3-3] and the ["Minimum Channels OPERABLE"] column in CTS Table [3.3-6] and the reference to them in the ACTIONS are relabeled as the "Required Channels" consistent with NUREG-1431. ACTION Statements are revised to use the ITS terminology, "Required Channels". Changing the column titles is purely administrative. The numbers in the columns are adjusted, if necessary. Where the numbers are adjusted, those changes are described in different CNS.
01-44	A	The "MODES For Which Surveillance Is Required" columns in CTS Tables 4.3-1 and 4.3-2 [] are deleted since this information is enveloped by CTS Tables 3.3-1 and [3.3-3] and is redundant given the integrated OPERABILITY/ SR format in improved TS Tables 3.3.1-1 and 3.3.2-1.
01-45	M	The Overtemperature [ΔT], Overpower [ΔT], Pressurizer Pressure - High, and Steam Generator Water Level - Low-Low trip functions, which currently reference ACTION Statement [6], are now referenced to new ACTION Statement [2.1], consistent with ITS 3.3.1 Condition E. This change is more restrictive since one less hour is available under new ACTION Statement [2.1] than under the combination of current ACTION Statement [6] and LCO 3.0.3.
01-46	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
01-47	A	A new note (f) is added and applied to the Functional Unit 6.c. The note is for clarification only as the CTS Table 3.3-1 indicates that only 1 channel is required to be OPERABLE and that there is no trip function under these conditions.
01-48	LS4	CTS ACTION 7 is revised to allow a reduction in power below P-9 in lieu of tripping the inoperable channel. The ACTION is also revised to allow bypassing a tripped channel for four hours for surveillance testing other channels. Note (j) is added to Table 3.3-1, Applicable Modes for Functional Unit 17.a and b that states that the requirements are only applicable above P-9.
01-49	LS18	CTS ACTION 9 is deleted and revised ACTION 6 is used which allows a power reduction below P-7 in lieu of tripping the inoperable channel. Note (g) is added that specifies that Functional Unit 19 of the CTS does not have to be applied until the power level associated with P-7 is reached. ACTION 6 also allows the tripped channel to be bypassed for up to 4 hours to perform surveillance testing on other channels.

CHANGE NUMBER

NSHC

DESCRIPTION

01-50	A	ACTION [28] of the CTS duplicates CTS ACTION [6] and is deleted.
01-51	LG	This change moves the description of the P-7 inputs, i.e., P-10 and P-13, to the Bases since they are duplicated by Functional Units [20.e and 20.f]. The Required Channels column for P-7 lists "1 per train" since this is a more appropriate convention for a logic function. These changes are consistent with NUREG-1431. [This change also deletes the surveillance requirements for P-7 per CN 3.3-54 in the ITS since the COTs and channel calibration apply to P-10 and P-13 not to the P-7 logic function.]
01-52	LG	This change moves the specifics on how to verify permissive functions of ACTIONS [8] and [21] to the Bases, consistent with NUREG-1431. This information is more appropriately controlled outside of the TS while the underlying requirement to verify proper permissive operation is unchanged.
01-53	A	CTS Table 3.3-1 ACTION Statement [2.c] is revised to be consistent with ITS SR 3.2.4.2, as discussed in CN 4-04-LS-12 in the 3/4.2 package.
01-54	LS37	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-55	LS39	APPLICABILITY Note [*] and ACTION Statement [11] for Functional Units [1, 6.b, 20, and 21] of CTS Table 3.3-1 are modified to provide an alternative to opening the reactor trip breakers (RTBs) while still assuring that the function and intent of opening the RTBs is met. As currently worded, these ACTION Statements result in a feedwater isolation signal (FWIS) when in MODE 3 with a T_{avg} less than [554°F. FSAR Table 7.3-3 and FSAR Figure 7.2-1 (sht. 13) detail the FWIS generation on the coincidence of P-4 and low T_{avg} .] A more generic action, which assures the rods are fully inserted and cannot be withdrawn, replaces the specific method of precluding rod withdrawal. The revised APPLICABILITY and ACTION Statements still assure rod withdrawal is precluded. This change does not involve any safety impact and is consistent with traveler TSTF-135.
01-56	A	The DCPP CTS 3.3.1 ACTION 2.c requires that power be reduced to less than 75% or that SR 4.2.4.2 be performed whenever power is $\geq 50\%$. This power level requirement should be $\geq 75\%$ since if power is decreased below 75% per the first part of Action 2.c, the required ACTION is complete and in addition, SR 4.2.4.2 is only required for power levels $\geq 75\%$ with one power range detector inoperable.

**CHANGE
NUMBER**

NSHC

DESCRIPTION

01-57	LG	CTS Table 3.3-1 Functional Units [12.a and 12.b] are combined per Traveler TSTF-169. The Required Channels, ACTION Statement, and Surveillance Requirements are the same for both Functional Units. The only difference between the two is the APPLICABILITY which could lead to entry into ACTION Statement 6 for Functional Unit [12.a], followed by a power reduction below P-8 exiting the APPLICABILITY and required ACTIONS for that Functional Unit, and subsequent re-entry into ACTION Statement 6 for Functional Unit [12.b]. This would involve an improper cumulative AOT of 12 hours before tripping an inoperable channel, beyond that evaluated in WCAP-10271 and its Supplements. The relationships between these Functional Units and permissives P-7 and P-8 are moved to the ITS 3.3.1 Bases.
01-58	A	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
01-59		Not Used.
01-60		Not Used.
01-61	M	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-01	A	The Engineered Safety Features Actuation System Instrumentation [Trip Setpoints and] Allowable Values are moved to ITS Table 3.3.2-1.
02-02	A	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-03	LG	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-04	LG	The requirements stipulated in ACTIONS a and b are moved to ITS Table 3.3.2-1, with explicit direction contained in the ITS ACTIONS Bases.
02-05	M	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-06	LS33	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).

CHANGE
NUMBER

NSHC

DESCRIPTION

02-07

LS11

[Note (a) is added to CTS Table 3.3-3 for the Steam Line Isolation Functional Units 4.a, 4.b, 4.c, 4.d, and 4.e to state that the LCO requirements are not applicable in MODES 2 and 3 when the MSIVs are closed and deactivated]. Note [(b)] is added to CTS Table [3.3-3] for the Feedwater Isolation and Turbine Trip Function [Functional Units 5.a, and 5.b] to state that the LCO requirements are not applicable when the [MFIVs, MFRVs or the associated bypass valves] are closed [and deactivated or isolated by a closed manual valve]. When these valves are closed [and deactivated or isolated by a closed manual valve], they are already performing their safety function. These changes are consistent with NUREG-1431.

02-08

M

[This change revises ACTION 20 and 35 in CTS Table 3.3-3 and adds new ACTION 20.2 and 35.2 which are applicable to Functional Units 1.c, 1.d, 1.ef, 4.c, 4.d, 4.e, 5.b, 6.dc.1)a, and 6.d]. These ACTION Statements, written to reflect the APPLICABILITY of the affected channels and consistency with ITS 3.3.2 [Conditions D and I], are more restrictive, by one hour, than the current ACTION Statement[s] which invoke[] LCO 3.0.3 if the inoperable channel is not placed in trip within 6 hours.

02-09

LG

Separate ESFAS entries for the motor-driven and turbine-driven auxiliary feedwater pumps are no longer necessary, consistent with NUREG-1431. The only difference in the requirements (an SR 4.0.4 exception for response time testing of the turbine-driven auxiliary feedwater pump) has been addressed in the ITS by a Note in Surveillance Requirement 3.3.2.10. [The details of which actuation signal starts which pump is moved to the Bases for SI and RCP undervoltage].

02-10

M

Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

02-11

A

The Functional Unit for Loss of Power [CTS 7.a, 7.b] is moved to improved TS 3.3.5.

02-12

M

Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

02-13

A

Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

02-14

M

This change modifies ACTION Statement [21] for permissive P-11 [] to provide specific shutdown requirements to exit APPLICABILITY in lieu of applying LCO 3.0.3. This change is more restrictive by one hour, consistent with NUREG-1431.

**CHANGE
NUMBER**

NSHC

DESCRIPTION

02-15	M	ACTION Statement [17] has been expanded to specify additional actions and options if an inoperable channel is not placed in bypass within the specified time period. In the CTS, this would have required an entry into LCO 3.0.3, which would necessitate that the plant initiate a shutdown within 1 hour and be in the next mode in 6 hours. In the ITS, the requirements to place the inoperable channel in bypass within a time constraint and the reduction, by 1 hour, in the time to exit APPLICABILITY are more restrictive than the CTS. [As a result of the revision to ACTION 17, a new ACTION 17.1 was created for Functional Unit 4.c that requires entry to MODE 4 if the required ACTIONS are not met.]
02-16	LS12	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-17	LS13	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-18	LS31	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-19	LG	This change moves functions provided by a Safety Injection signal and the AFW pump start entries [] for Functional Units [6.d, and 6.e] in CTS Table [3.3-3] to the ITS 3.3.2 Bases, consistent with NUREG-1431.
02-20	A	The Functional Unit for Containment Ventilation Isolation is moved to ITS 3.3.6. There are no technical differences introduced by this process[].
02-21	LS22	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-22		Not Used.
02-23	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-24	LS19	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-25	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-26	LS21	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-27	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
02-28	LG	This change moves information inserted by LA 114/112 on containment spray and safety injection coincidence to the Bases, consistent with NUREG-1431.
02-29	M	A new functional unit 9 is added, per a License Amendment Request, that incorporates ACTION 20.1(new) and Surveillances for the residual heat removal (RHR) pump trip from low refueling water storage tank level.
02-30	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-31	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-32	LS23	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-33	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-34	LS34	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-35	A	Note (2) of the CTS Table 4.3.2 is revised. The testing frequency was relaxed from monthly (M) to quarterly (Q) via License Amendment 102/101, but the Note was not revised nor was it shown as applicable to Functional Unit 3.c.4).
02-36	M	This change revises the APPLICABILITY of Functional Unit 7 to require OPERABILITY when the associated DG is required to be OPERABLE by LCO 3.8.2 of the ITS. This change is consistent with NUREG-1431.
02-37	LG	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-38	LS35	Delete MODE 4 APPLICABILITY from the Manual Initiation of MSIVs since the valves are not required to be OPERABLE in MODE 4 per CTS 3.7.1.5 or ITS 3.7.2.
02-39	LG	Move valve numbers in CTS ACTION 18 dealing with containment ventilation isolation to the Bases, consistent with NUREG-1431.
02-40	A	This administrative change affects the manner in which the CTS 4.0.4 exception for testing the TDAFW pump is presented. The exception allows entry into MODE 3 to perform the TDAFW pump response time testing. In NUREG-1431, the CTS 4.0.4 exception from [CTS 3.7.1.2 has been interpreted so that it allows response time testing to be deferred as] is reflected in the ITS SR 3.3.2.10 NOTE.

**CHANGE
NUMBER**

NSHC

DESCRIPTION

02-41	LS36	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-42	LS38	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-43	LG	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-44	A	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-45	LG	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-46	LS42	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-47	M	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-48	LS28	A new Action 15 is added and applied to Functional Unit 7. a. 2) and 7. b. 1) that allows the affected Emergency Diesel Generator to be declared inoperable and requires entry into Specification 3.8.1.1 when more than one relay per bus is inoperable. Current ACTION 16 does not address the above situation and requires entry into LCO 3.0.3. This change is consistent with NUREG-1431.
03-01	A	The requirements of this specification [CTS 3.3.3.1] are moved to [four] separate specifications in the improved TS. The RCS Leakage Detection requirements are moved to improved TS 3.4.15. [The Fuel Building requirements are moved to improved TS 3.3.8.] The Control Room requirements are moved to improved TS 3.3.7. [The Containment Ventilation Isolation requirements are moved to improved TS 3.3.6.]
03-02	M	The requirements stipulated in ACTION [a] are moved to ITS Tables [3.3.6-1, 3.3.7-1 and 3.3.8-1], with explicit direction contained in the ITS ACTIONS Bases. The 4 hour AOT for setpoint adjustment is eliminated.
03-03	LG	The requirements associated with the criticality monitors are moved to a licensee controlled document. These monitors are required by 10CFR70.24; however, there is no requirement for [them] to be in the Technical Specifications [as criticality monitors. They are retained, however, as initiators of the Iodine Removal mode of the FHBVS for a fuel handling accident until RM-44A and 44B are installed in accordance with License Amendment 70/69]. Since Part 70 is invoked in the operating license, these monitors will be retained in the plant design.

**CHANGE
NUMBER**

NSHC

DESCRIPTION

03-04	M	This change adds the APPLICABILITY for movement of irradiated fuel assemblies consistent with NUREG-1431. The CTS APPLICABILITY of "All" MODES does not cover the movement of irradiated fuel assemblies when the core is offloaded.
03-05	LS14	ACTION Statement [34] for the [Control Room Air Intake] [] radiation monitors have extended Completion Times, from [1 hour] to 7 days for one required channel inoperable, consistent with NUREG-1431.
03-06	A	ACTION [c] of CTS LCO 3.3.3.1 is revised to state the Specification 3.0.3 exception is [retained only for the Fuel Handling Building Radioactivity Instrumentation]. The LCO 3.0.3 exception is not needed in ITS 3.3.7 or ITS 3.4.15 since Required Actions are provided with the appropriate remedial measures for all combinations of failures, including shutdown actions, or reference is made to the associated plant system TS for the systems affected by the inOperability of the radiation monitors. [].
03-07	LS16	The APPLICABILITY for the Fuel Building Exhaust radiation monitors has been revised to read "during movement of irradiated fuel assemblies in the fuel [handling] building." [The REQUIRED CHANNELS for Instrument 1.b. has been revised from one as specified by the CTS to two as specified by NUREG-1431 to provide protection against a single failure that could prevent the transfer of the FHBVS to the iodine removal mode.]
03-08	M	The CTS have been revised to include manual initiation of the fuel handling building and manual and automatic initiation of the control room pressurization system. These systems are not classified as ESF functions in the CTS even though CTS surveillance 4.7.5.1e.2) requires that the CRVS automatically switches to the pressurization mode on a Phase "A" signal. The FHBVS is not an ESF function since its only function is to mitigate a fuel handling accident. This revision incorporates the Actuation Logic, Master Relay, and Slave Relay Tests included in NUREG-1431 for the CRVS and the TADOT for the manual actuation of both systems. The automatic actuation tests are conducted as part of the CTS, and the relay tests are currently performed even though not specifically called out in the CTS.
03-09	LS-24	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
03-10	LG	The DCP descriptive information related to the Required Channels per normal intake is moved to the Bases.

CHANGE NUMBER

NSHC

DESCRIPTION

03-11	M	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
03-12	A	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
03-13	M	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
03-14	LS29	This proposed change adds an ACTION and an allowed outage time of 4 hours for one inoperable Containment Ventilation Radiation instrumentation or actuation channel. The CTS via ACTIONS 18 and 33 requires that for one or two instruments or channels inoperable that CTS 3.6.3 or 3.9.9 be entered. The revised TS will require that ITS 3.6.3 or 3.9.4 be entered if the instrument or channel cannot be returned to an OPERABLE status within the revised AOT. This change is consistent with the requirements of NUREG-1431.
03-15	M	This change revises CTS ACTION 34 to require appropriate MODE changes or condition changes for the CRVS with one inoperable normal intake monitor and new ACTION 36 specifies actions for two inoperable normal intake monitors. The CTS requires that if the required ACTIONS for one inoperable CRVS monitor is not met that LCO 3.0.3 be entered. In addition, the CTS does not specify a required action if both monitors are inoperable. NUREG-1431 requires that for the above conditions that appropriate actions be taken to place the plant in a condition of non-APPLICABILITY. These ACTIONS specify a shutdown requirement for MODES 1-4 that is one hour less than LCO 3.0.3, and immediate ACTION for inoperability in MODE 5 or 6, and immediate action for inoperability during fuel movement. These changes are consistent with NUREG-1431. Refer also to CN 03-08-M, CN 03-04-M, and CN 3.3-51.
03-16	A	NOT USED ITS 3.3.6 for DCPD includes MODES 1-4 and during movement of irradiated fuel assemblies within containment, in addition to MODE 6, in the LCO APPLICABILITY. These requirements are inferred in CTS 3.6.3 and are repeated here for clarity.
03-17	A	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
04-01	R	DCPD LCO 3.3.3.2, Movable Incore Detectors, is relocated to a licensee controlled document, see Attachment 21, page 11.
05-01	R	DCPD LCO 3.3.3.3, Seismic Instrumentation, is relocated to a licensee controlled document, see LAR 95-07.

**CHANGE
NUMBER**

NSHC

DESCRIPTION

06-01	R	DCPP LCO 3.3.3.4, Meterological Instrumentation, is relocated to a licensee controlled document, see Attachment 21, page 13.
07-01	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-03		Not Used.
07-04	LS15	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-05	A	Consistent with the ITS, the modifications would clarify the requirement to be in HOT SHUTDOWN in 12 hours by replacing the requirement with a new requirement to be in HOT STANDBY in 6 hours and in HOT SHUTDOWN in the next 6 hours.
07-06	LG	The Readout Location and Total No. of Channels columns in CTS Table [3.3-9] have been moved to the Bases of improved TS 3.3.4. [Descriptive information related to the controls is also moved to the Bases.]
07-07		Not Used.
07-08	TR2	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-09	LS43	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-10	LS26	In the CTS, the determination that the steam generators are available for decay heat removal, using instrumentation available at the remote shutdown panel, is based on steam generator level and auxiliary feedwater flow to the steam generator. In the ITS, it is recognized that either of these indications is sufficient.
07-11	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-12	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-01	A	This change, consistent with NUREG-1431, revises "Channel" and "Instrument" to "Function."
08-02		Not Used.

CHANGE
NUMBER

NSHC

DESCRIPTION

08-03	A	This change revises CTS Table [3.3-10] to clarify the number of channels required to be Operable. This is an administrative change which deletes the "Minimum Channels Operable" column []. The required ACTIONS are now based on one channel inoperable or two channels inoperable, rather than "less than the Total Number" or "less than Minimum Number." This change is consistent with NUREG-1431.
08-04	LS17	Consistent with NUREG-1431 (ITS 3.3.3 Required ACTIONS C.1, E.1, and G.1), this change deletes the requirement to initiate an alternate means of monitoring within 72 hours when two channels of Containment Radiation Level [or RVLIS] are inoperable as specified in CTS [3.3.3.6 ACTION d. In addition, a special report is required within 14 days that identifies the alternate method of monitoring the appropriate parameter(s), as well as the current special report requirements].
08-05	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
08-06	LG	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
08-07	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
08-08	LS27	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
08-09		Not Used.
08-10		Not Used.
08-11	LS30	This change revises the DCCP CTS 3.3.3.6 to conform to NUREG-1431 and revises CTS Table 3.3-10 to both add and delete instruments per the Reviewer's Note on ISTS Table 3.3.3-1.
09-01	LG	The explosive gas monitoring instrumentation will be controlled by the Explosive Gas Monitoring Program established in accordance with ITS 5.5.12, see Attachment 21, page 15.
10-01	R	The Turbine Overspeed Protection System is relocated to a licensee controlled document, see LAR 95-07.
11-01	R	LCO 3.3.3.7, Chlorine Detection Systems, is relocated to a licensee controlled document, see LAR 95-07.

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(31 pages)

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	A Note, "Separate Condition entry is allowed for each Function," is added to the ACTIONS for the Reactor Trip System, ESFAS, [] and Accident Monitoring Instrumentation. This change clarifies those situations where the CTS ACTION Statements are not uniquely associated with a particular Function or where the required channels are specified on a per steam line, per loop, per SG, per bus, etc., basis.	Yes	Yes	Yes, see also CN 2-46-LS-42.	Yes, see also CN 2-46-LS-42.
01-02 LG	The improved TS specify that required response time testing be performed on a STAGGERED TEST BASIS and do not impose any requirements as to which train should be tested. The requirement to ensure that each train is tested every 36 months is moved to the Bases for SR 3.3.1.16 and SR 3.3.2.10.	Yes	Yes	Yes	Yes
01-03 LS1	Changing "demonstrated" to "verified" allows Reactor Trip System and ESFAS sensor response time verifications to be performed per WCAP-13632-P-A Revision 2. This change is consistent with traveler TSTF-111.	Yes	No, see CN 1-58-A.	Yes	Yes
01-04 LG	In CTS Tables [3.3-1 and 3.3-3], the ["Channels to Trip" and] "Minimum Channels OPERABLE" columns are deleted. [] The ACTION Statements have been revised accordingly.	Yes	Yes	Yes	Yes
01-05 A	The LCO 3.0.4 exception [footnote #] in CTS Table 3.3-1 is deleted entirely. ACTION Statement [8] in CTS Table 3.3-1 permits continued operation for an unlimited period of time. Therefore, no exception to ITS LCO 3.0.4 is needed for this ACTION Statement. []	Yes	No, not in CTS.	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-06 LS2	A new ACTION Statement [2.1] is created which does not require a reduction in THERMAL POWER to less than 75% RTP or the measurement of the QPTR if above 75% RTP.	Yes	Yes	Yes	Yes
01-07 LS3	<p>With one intermediate range neutron flux channel inoperable when the plant is above P-6 but below 10% RTP (the P-10 interlock setpoint), current ACTION Statement [3.b] is revised to establish a 24 hour Completion Time for channel restoration or changing the power level to either below P-6 or above P-10. Although this change is less restrictive since a power increase is an allowed option, the ACTION Statement would ensure protection by entering the range of the four power range neutron flux channels. The Applicability for Functional Unit 5 is revised such that current ACTION Statement [3.a] can be deleted.</p> <p>With both intermediate range neutron flux channels inoperable when the plant is above P-6 but below P-10, LCO 3.0.3 would be entered under the CTS and the plant would have to be in MODE 3 within 7 hours. New ACTION Statement [3.1] requires immediate suspension of operations involving positive reactivity additions and a power reduction below P-6 within 2 hours.</p>	Yes	Yes	Yes	Yes
01-08 M	Current ACTION Statement [4], for one source range channel inoperable in MODE 2 below P-6, is revised to require "immediate" suspension of operations involving positive reactivity changes and a new ACTION Statement [4.1] is added to address the condition where both source range channels are inoperable in MODE 2 below P-6 [] to require immediate opening of the RTBs.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-09 M	This DCPD-specific change revises CTS ACTION 5 to require the suspension of positive reactivity additions. Reference to specification 3.1.1.2 is deleted since the ITS reference is only to 3.1.1.1 which is more conservative for MODE 5.	Yes	No	No	No
01-10 LS32	<p>This Callaway-specific change redefines the Actions for the source range neutron flux channels in MODES 3, 4, and 5. The source range neutron flux channels are required to provide input to the Reactor Trip System to mitigate potential uncontrolled rod withdrawal events only when the Rod Control System is capable of rod withdrawal or all rods are not fully inserted (see CN 1-55-LS-39).</p> <p>If one source range neutron flux channel is inoperable in MODES 3, 4, and 5 when rod motion is possible, ACTION Statement 5.a is entered. If the channel is not restored within 48 hours, the Applicability is exited by inserting all rods and precluding rod withdrawal. If both source range channels are inoperable when rod motion is possible, new ACTION Statement 4.1 is entered and the reactor trip breakers are immediately opened.</p> <p>The current Action Statement 5.a requirements regarding the suspension of positive reactivity changes and closure of dilution source valves are deleted since they are not related to the reactor trip function.</p>	No	No	No	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-11 LS5	[New] ACTION Statement [39] is applied to the Low Fluid Oil Pressure and Turbine Stop Valve Closure trip functions. Rather than entry into LCO 3.0.3 if current ACTION Statements [6 and 11] are not met or if multiple low fluid oil pressure channels are inoperable, the new ACTION Statement requires inoperable channels to be tripped within 6 hours or power reduced below P-9 within 10 hours.	No, see CN 1-48-LS-4.	Yes	Yes	Yes
01-12 M	New ACTION Statement [8.1] is created to differentiate between those RTS interlocks required to be operable in MODE 1 only, and those interlocks required to be operable in MODES 1 and 2. If the interlock function is required to be operable in MODE 1 only and the LCO and ACTION requirements are not met, then new ACTION Statement [8.1] requires that the unit be taken to MODE 2 within 7 hours. In addition, current ACTION Statement [8] is revised for those interlocks required to be OPERABLE in MODES 1 and 2. If one channel is inoperable, the interlock must be determined to be in its required state or the plant must be in at least HOT STANDBY within 7 hours.	Yes	Yes	Yes	Yes
01-13 LS6	[ACTION Statement [10] is revised to note that the 2 hour [train and] reactor trip breaker bypass allowance for [train or] breaker surveillance testing can also be used for maintenance.] ACTION Statement [10] is [also] revised to require restoration of an inoperable RTB within 1 hour or the plant must be in HOT STANDBY within the next 6 hours. This is less restrictive since an additional hour is provided for the transition to MODE 3.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-14 A	In the ISTS Table 3.3.1-1, Function 20, the Reactor Trip Breaker (RTB) Undervoltage and Shunt Trip Mechanisms are separate from the RTB Functional Unit. The CTS have been revised to reflect these requirements. New [footnote (k) has] been added to the RTB Functional Unit to note that the same OPERABILITY requirements and ACTIONS apply to a bypass breaker if it is racked in and closed for bypassing an RTB. The bypass breakers were already handled in this fashion. ACTION Statement [12] in CTS Table 3.3-1 has been revised accordingly.	Yes	Yes	Yes	Yes
01-15 A	The Applicability for the Reactor Trip on Turbine Trip function is modified by new footnote [(c)] such that this function is only required to be OPERABLE above the P-9 interlock setpoint (50% RTP). This is acceptable since the trip function is blocked below P-9. [New] ACTION Statement [39] is applied to the Low Fluid Oil Pressure and Turbine Stop Valve Closure trip functions.	No, see CN 1-48-LS-4.	No, already in CTS.	Yes	Yes
01-16 LS40	The requirement to verify the setpoint during the quarterly TADOT for RCP Underfrequency [and RCP Undervoltage] is deleted.	Yes	Yes	Yes	Yes
01-17 A	The bypass allowance can be used for surveillance testing or setpoint adjustment. Setpoint adjustment can be performed at power and may be required by other Technical Specifications.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-18 LS7	The CTS requirement to reduce the Power Range Neutron Flux Trip setpoints in the event a power range flux channel is inoperable is deleted. The time to reduce power below 75% RTP is increased from 4 hours to 12 hours and, if actions are not completed as required, the unit must be in MODE 3 in 12 hours. (See also CN 1-53-A.)	Yes	Yes	Yes	Yes
01-19 LS8	If the requirements of current ACTION Statement [6] are not met, LCO 3.0.3 would be entered. In accordance with the ISTS, this ACTION Statement is revised to state that, if the ACTION requirements are not met, the plant must be taken below the P-7 interlock setpoint within the next 6 hours. [The Applicability for Functional Units 9, 11, 12, 15, and 16 in CTS Table 3.3-1 is also revised to add new footnote (g).]	Yes	No, see CN 1-61-M.	Yes	Yes
01-20 A	Callaway's current ACTION Statement 31 is reformatted per NUREG-1431 Rev. 1 to require restoration of an inoperable channel within 6 hours or the plant must be taken to MODE 3 within 12 hours. This is an administrative change since the total time to exit the Applicability is unchanged.	No	No	No	Yes
01-21 A	This change reflects the reorganization of the surveillances on the incore/excore axial flux difference. There is no change to the surveillances or how they are performed. See also CN 1-25-A.	Yes	Yes	Yes	Yes
01-22 M	Quarterly COTs have been added for power range - low and intermediate range flux channels. [The requirement to verify the state of P-6 and P-10 has been added for these COTs.]	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-23 A	This change adds notes to the RTS and ESFAS SR Tables 4.3-1 and 4.3-2 that explicitly require the 18-month calibrations to include verifications of affected time constants where applicable.	Yes	Yes	Yes	Yes
01-24 LS9	The COTs for the Power Range Neutron Flux - Low setpoint, the Intermediate Range Neutron Flux and the Source Range Neutron Flux trip functions will no longer be required if performed within the previous 92 days (extended from 31 days). Note [1a] is added for use with the turbine trip functions, for which no such change was provided.	Yes	Yes	Yes	Yes
01-25 A	NUREG-1431 Rev. 1 incorporates the CTS 4.0.4 exception from Table 4.3-1 Notes (2), (3), and (6) into the ITS SR 3.3.1.2, 3.3.1.3, and 3.3.1.6 surveillance frequencies.	Yes	Yes	Yes	Yes
01-26 LG	This change moves detail concerning NIS detector operation and testing to the BASES for ITS SR 3.3.1.11.	Yes	Yes	Yes	Yes
01-27 LS10	Surveillances on the Source Range Neutron Flux trip function are reorganized to reflect plant status in accordance with NUREG-1431. New Note [(19)] requires that the quarterly COT be performed within 4 hours after reducing power below the respective source range instrumentation Applicabilities, if not performed within the previous [92] days. Since the COT is valid for [92] days, there is no need to repeat it if one has been performed within the prior [quarter]. Since the CTS has no Specification 4.0.4 exception, this 4 hour allowance is less restrictive.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-28 A	Note [8] is revised to require the P-6 and P-10 interlock verification to be performed during all source range COTs. These permissives are verified to be in their correct state prior to entry into MODES 3, 4, and 5 during shutdown and after leaving MODES 3, 4, and 5 during startup.	Yes	Yes	Yes	Yes
01-29 LG	This change moves the details regarding measurement of loop-specific ΔT values to the BASES for ITS SR 3.3.1.6.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
01-30 M	Boron Dilution Mitigation System (BDMS) signal blocking and surveillance requirements are moved to ITS LCO 3.3.9. This is a more restrictive requirement since the BDMS, other than the inputs from the source range channels, currently has no LCO or ACTION requirements.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
01-31 A	One-time surveillance waivers are deleted. They are no longer applicable.	No, not in CTS.	Yes	Yes	Yes
01-32 LG	[] [Note (10) of CTS Table 4.3-1 is deleted since it is redundant; every TADOT requires independent UVTA and STA verification per ITS SR 3.3.1.4, not just those TADOTs following maintenance or adjustment.] Notes [(14) and (16)] of CTS Table 4.3-1 are moved to the BASES for ITS SR 3.3.1.14.	Yes	Yes	Yes	Yes
01-33 TR1	The BDMS actuation SR is changed to allow the use of an actual signal, if and when one occurs, to satisfy surveillance requirements.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-34 A	Callaway's current ACTION Statement 5.b requirements for two inoperable source range channels are divided between the reactor trip, indication, and BDMS functions served by the source range channels. New ACTION Statement 4.1, added to CTS Table 3.3-1, retains the requirement to open the reactor trip breakers to serve the reactor trip function. The ACTION Statement 5.b requirements that are not related to reactor trip are moved to ITS 3.3.9. See also CN 1-30-M.	No	No	No	Yes
01-35 LG	This change moves response time limit tables to the updated FSAR per Generic Letter 93-08 and NUREG-1431 Rev. 1.	Yes	No, already moved to TRM.	No, already moved to USAR Section 16.3.	No, already moved to FSAR Section 16.3.
01-36 M	This DCCP-specific change adds a requirement to perform a Channel Calibration on Functional Unit 17 every 18 months.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-37 A	In the CTS, the "Minimum Channels OPERABLE" is one less than the "Total Number of Channels" for Functional Units [22.c] (P-8), [22.d](P-9), and [22.e](P-10) in Table 3.3-1 and Functional Unit [8.a](P-11) in Table [3.3-3]. For these Reactor Trip System and ESFAS interlocks, current ACTION Statements [8 and 21] for an inoperable channel are based on the "Minimum Channels OPERABLE" columns in Tables 3.3-1 and [3.3-3]. In the improved TS, only the "Total Number of Channels" information is retained in the LCO and that column is relabeled as the "Required Channels", as discussed in CN 1-04-LG and CN 1-43-A. Required Actions in improved TS 3.3.1 Conditions S and T and improved TS 3.3.2 Condition L are tied to the Required Channels. Therefore, the required permissive channels for these Functional Units are revised in the CTS. Refer also to CN 1-51-LG for P-7.	Yes	Yes	Yes	Yes
01-38 R	The source range channel operability requirements in MODES 3, 4, and 5 when incapable of rod withdrawal are relocated to a licensee controlled document.	No, see CN 1-09-M.	Yes, relocated to the TRM.	Yes, relocated to Chapter 16.3 of the USAR.	No, see Cns 1-10-LS-32, 1-32-M, and 1-34-A.
01-39 A	This change adds Note [(5)] to Functional Units [2.b, 3, and 4] in CTS Table 4.3-1.	Yes	Yes	Yes	Yes
01-40 LS41	Surveillance intervals applied to Notes (3) and (6) of CTS Table 4.3-1 will be defined in terms of effective full power days (EFPD). However, EFPD may be longer than calendar days as specified in the CTS. Therefore, this change is considered less restrictive.	No, already in CTS.	No, already in CTS.	Yes	Yes
01-41 A	This DCCP-specific change moves the first sentence of note (1) of Table 3.3-2 to ITS SR 3.3.1.16, and moves the rest of note (1) to the Bases.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-42 M	This change adds direction to ACTIONS 7, 11, and 13 in CTS Table 3.3-1 to be in MODE 3 within 12 hours, in lieu of LCO 3.0.3 entry, if inoperable EAM/TTD timer(s) or channel(s) aren't tripped within 6 hours.	No, not in current design or TS.	No, not in current design or TS.	No, not in current design or TS.	Yes
01-43 A	The "Total Number of Channels" columns in CTS Tables 3.3-1 and [3.3-3] and the ["Minimum Channels OPERABLE"] column in CTS Table [3.3-6] and the references to them in the ACTIONS are relabeled as the "Required Channels" consistent with NUREG-1431 Rev. 1. ACTION Statements have been revised accordingly.	Yes	Yes	Yes	Yes
01-44 A	This change deletes the "MODES For Which Surveillance Is Required" column in CTS Tables 4.3-1[, 4.3-2 and 4.3-3].	Yes	Yes	Yes	Yes
01-45 M	The Overtemperature [ΔT], Overpower [ΔT], Pressurizer Pressure - High, and Steam Generator Water Level - Low-Low trip functions, which currently reference ACTION Statement [6], are now referenced to new ACTION Statement [2.1], consistent with ITS 3.3.1 Condition E. This change is more restrictive since one less hour is available under new ACTION Statement [2.1] than under the combination of current ACTION Statement [6] and LCO 3.0.3.	Yes	Yes	Yes	Yes
01-46 A	ACTION Statement 13 of CTS Table 3.3-1 and ACTION Statement 36 of CTS Table 3.3-3 are revised to reflect operating and testing options that have existed since the SG Water Level Low-Low EAM/TTD design was implemented, but were not listed in the TS since they were not necessarily the options of choice.	No, not in current design or TS.	No, not in current design or TS.	No, not in current design or TS	Yes, reviewed in OL Amendment No. 43 dated April 14, 1989.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-47 A	For DCP, a new note (f) is added and applied to the Functional Unit 6.c. The note is for clarification only as the CTS Table 3.3-1 indicates that only 1 channel is required to be OPERABLE and that there is no trip function under these conditions.	Yes	No	No	No
01-48 LS4	For DCP, CTS ACTION 7 is revised to allow a reduction in power below P-9 in lieu of tripping the inoperable channel. The action is also revised to allow bypassing a tripped channel for four hours for surveillance testing other channels. Note (j) is added to Table 3.3-1, Applicable Modes for Functional Unit 17.a and b that states that the requirements are only applicable above P-9.	Yes	No	No	No
01-49 LS18	For DCP, CTS ACTION 9 is deleted and revised ACTION 6 is used which allows a power reduction below P-7 in lieu of tripping the inoperable channel. Note (g) is added that specifies that Functional Unit 19 of the CTS does not have to be applied until the power level associated with P-7 is reached. ACTION 6 also allows the tripped channel to be bypassed for up to 4 hours to perform surveillance testing on other channels.	Yes	No	No	No
01-50 A	ACTION [28] of the CTS duplicates CTS ACTION [6] and is deleted.	Yes	Yes	No, not in CTS.	No, not in current TS.
01-51 LG	This change moves the description of the P-7 inputs, i.e., P-10 and P-13, to the Bases since they are duplicated by Functional Units [22.e and 22.f] and lists "1 per train" under the Required Channels column. [This change also deletes the surveillance requirements for P-7 per CN 3.3-54 in the ITS since the COTs and channel calibration apply to P-10 and P-13 not to P-7 logic function.]	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-52 LG	This change moves the specifics on how to verify permissive functions of ACTIONS [8] and [21] to the Bases.	Yes	Yes	Yes	Yes
01-53 A	CTS Table 3.3-1 ACTION Statement [2.c] is revised to be consistent with ITS SR 3.2.4.2, as discussed in CN 4-04-LS-12 in the 3/4.2 package.	Yes	Yes	Yes	Yes
01-54 LS37	ACTION Statement 5.b of Callaway's CTS Table 3.3-1 is revised to change the 14 day recurring verification of the closed status of the unborated water source isolation valves to 31 days.	No	No	No	Yes
01-55 LS39	Applicability Note [*] and ACTION Statement [11] for Functional Units [1, 6.b, 20, and 21] of CTS Table 3.3-1 are modified to provide an alternative to opening the reactor trip breakers (RTBs) while still assuring that the function and intent of opening the RTBs is met.	Yes	Yes	Yes	Yes
01-56 A	The DCPD CTS 3.3.1 Action 2.c requires that power be reduced to less than 75% or that SR 4.2.4.2 be performed whenever power is $\geq 50\%$. This power level requirement should be $\geq 75\%$ since if power is decreased below 75% per the first part of Action 2.c, the required Action is complete and in addition, SR 4.2.4.2 is only required for power levels $\geq 75\%$ with one power range detector inoperable.	Yes	No	No	No
01-57 LG	CTS Table 3.3-1 Functional Units [12.a and 12.b] are combined per TSTF-169. The relationship between Functional Units is moved to the Bases.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-58 A	The proposed change would allow Reactor Trip System and ESFAS sensor response time testing to be performed per WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," or other similar methodologies. This change is consistent with traveler TSTF-111, which revises the Bases for ITS SR 3.3.1.16 and SR 3.3.2.10 to allow the elimination of pressure sensor response time testing.	No, see CN 1-03-LS1.	Yes	No, see CN 1-03-LS1.	No, see CN 1-03-LS1.
01-59	Not Used.	N/A	N/A	N/A	N/A
01-60	Not Used.	N/A	N/A	N/A	N/A
01-61 M	If the requirements of current CPSES ACTION Statement 6 are not met, LCO 3.0.3 would be entered. In accordance with the ISTS, this ACTION Statement is revised to state that, if the ACTION requirements are not met, the plant must be taken below the P-7 interlock setpoint within the next 6 hours.	No, see CN-01-19-LS8.	Yes	No, see CN-01-19-LS8.	No, see CN-01-19-LS8.
02-01 A	The Engineered Safety Features Actuation System Instrumentation [Trip Setpoints and] Allowable Values are moved to ITS Table 3.3.2-1.	Yes	Yes	Yes	Yes
02-02 A	CTS ACTION b.1, Equation 2.2-1, and the values for Total Allowance (TA), Z, and Sensor Error (S) are deleted, consistent with NUREG-1431 Rev. 1.	No, not in CTS.	No, not in CTS.	Yes	Yes
02-03 LG	The Engineered Safety Features Actuation System Instrumentation Trip Setpoints are moved to a licensee controlled document.	No, retained in ITS.	Yes, moved to Bases.	Yes, moved to ITS 3.3.2 Bases.	Yes, moved to ITS 3.3.2 Bases.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-04 LG	The requirements stipulated in ACTIONS a and b are moved to ITS Table 3.3.2-1, with explicit direction contained in the ITS ACTIONS Bases.	Yes	Yes	Yes	Yes
02-05 M	The Functional Unit for Containment Purge Isolation, CTS 3.c, is moved to improved TS 3.3.6. Improved TS 3.3.6 adds requirements on the OPERABILITY of the containment purge radiation monitors and extends the Applicability to the Manual Initiation and BOP ESFAS actuation logic to include during movement of irradiated fuel assemblies within containment and Core Alterations.	No, see CN 02-20-A.	No, see CN 2-20-A.	Yes	Yes
02-06 LS33	Functional Unit 4.a.1 of curret TS Table 3.3-3 has been deleted.	No, retained in CTS.	No, see CN 2-25-A.	No, see CN 2-25-A.	Yes
02-07 LS11	[Note (a) is added to CTS Table 3.3-3 for the Steam Line Isolation Function to state that the LCO requirements are not applicable in MODES 2 and 3 when the MSIVs are closed and deactivated]. Note [(b)] is added to CTS Table [3.3-3] for the Feedwater Isolation and Turbine Trip Function to state that the LCO requirements are not applicable when the [MFIVs, MFRVs and the associated bypass valves] are closed [and deactivated or isolated by a closed manual valve].	Yes	Yes	Yes	Yes
02-08 M	[This change revises ACTION 20 and 35 in CTS Table 3.3-3 and adds new ACTION 35.2 which are applicable to Units 1.c, 1.d, 1.f, 4.d, 4.e, 5.b, 6.c.2.a, and 6.d]. These ACTION Statements, written to reflect the Applicability of the affected channels, are more restrictive, by one hour, than the current ACTION Statement[s] which invoke[] LCO 3.0.3 if the inoperable channel is not placed in trip within 6 hours.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-09 LG	Separate ESFAS entries for the motor-driven and turbine-driven auxiliary feedwater pumps are no longer necessary. The only difference in the requirements (an SR 4.0.4 exception for response time testing of the turbine-driven auxiliary feedwater pump) has been addressed in the ITS by a Note in Surveillance Requirement 3.3.2.10. [The details of which actuation signal starts which pump is moved to the Bases for SI and RCP undervoltage.]	Yes, moved to Bases.			
02-10 M	New ACTION Statement [19.1] is added to CTS Table [3.3-3] for one inoperable Main Feedwater Pump trip - AFW start channel. This is a more restrictive change since it reduces by 1 hour the Completion Time to reach MODE 3 as compared with entry into LCO 3.0.3.	No, not in CTS.	Yes	Yes	Yes
02-11 A	The Functional Unit for Loss of Power [CTS 7.a, 7.b] is moved to improved TS 3.3.5.	Yes	Yes	Yes	Yes
02-12 M	This Functional Unit [CTS 9.a-9.d] is moved to improved TS 3.3.7. The Applicable MODES have been expanded to include movement of irradiated fuel assemblies to cover fuel handling accidents.	No, not in CTS.	Yes	Yes	Yes
02-13 A	This Functional Unit [CTS 10] is moved to ITS 3.8.1.	No, not in CTS.	Yes	Yes	Yes
02-14 M	This change modifies ACTION Statement [21] for permissive P-11 [] to provide specific shutdown requirements to exit Applicability, in lieu of applying LCO 3.0.3. This change is more restrictive by one hour.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-15 M	Action Statement [17] has been expanded to specify additional actions and options if an inoperable channel is not placed in bypass within the specified time period. In the ITS, the requirements to place the inoperable channel in bypass within a time constraint and the reduction, by 1 hour, in the time to exit Applicability are more restrictive than the CTS. [As a result of the revision to ACTION 17, a new ACTION 17.1 was created for Functional Unit 4.c that requires entry to MODE 4 if the ACTIONS are not met.]	Yes	Yes	Yes	Yes
02-16 LS12	ACTION Statement [26] for an inoperable channel in the CTS Table [3.3-3] Functional Unit [9.a-9.c] is modified to be consistent with NUREG-1431 Rev. 1 (7 day AOT in ITS 3.3.7).	No, not in CTS.	Yes	Yes	Yes
02-17 LS13	The monthly TADOT has been extended to quarterly.	No, not in CTS.	Yes	No, retained CTS.	No, retained CTS.
02-18 LS31	CTS Table 3.3-3 ACTION Statement 19 is revised to reflect ITS 3.3.5 for the Loss of Power Functional Unit.	No, see CN 2-48-LS28.	No, see CN 2-32-LS-23.	Yes	Yes
02-19 LG	This change moves functions provided by a Safety Injection signal and the AFW pump start entries [] for Functional Units [6.d, and 6.e] in CTS Table [3.3-3] to the ITS 3.3.2 Bases.	Yes	Yes	Yes	Yes
02-20 A	The Functional Unit for Containment Vent Isolation is moved to ITS 3.3.6. There are no technical differences introduced by this process[].	Yes	Yes	No, see CN 2-05-M.	No, see CN 2-05-M.
02-21 LS22	In CPSES CTS Table 3.3-2, Action 17.1 replaces Action 17 for RWST Level Low-Low and Action 17.2 replaces Action 17 for SG Water Level - High High.	No	Yes	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-22	Not Used.	N/A	N/A	N/A	N/A
02-23 A	This change revises ACTIONS [14, 27, 27a, 34, and 34a] in CTS Table 3.3-3 to clarify that the 12 hour AOT to get to MODE 3 includes 6 hours for restoration followed by a 6 hour shutdown to MODE 3.	No, already in CTS.	No, already in CTS.	Yes	Yes
02-24 LS19	At CPSES, the Required Action for an inoperable SI sequencer is revised per ITS 3.8.1 Condition F to extend the time available for restoration of an inoperable SI sequencer from 6 hours to 12 hours.	No	Yes	No	No
02-25 A	The requirement for manual actuation of a single main steam isolation valve has been moved from CTS Table [3.3-3]. Operability of individual lines is addressed under ITS LCO 3.7.2.	No, retained CTS.	Yes	Yes	Yes
02-26 LS21	The Required Action for an inoperable Control Room [Isolation] [Manual Initiation, SSPS, or BOP-ESFAS] channel is modified to provide appropriate actions with the number of OPERABLE channels [two] less than the number of Required Channels.	No, not in CTS.	Yes	Yes	Yes
02-27 A	The CPSES Action Statement for the loss of offsite power - start motor driven auxiliary feedwater pumps is modified to require that if the Action Statements are not satisfied, the plant be taken to MODE 4 to exit the LCO Applicability. The CTS requires that MODE 5 be entered; however, because the function is only applicable in MODES 1, 2, and 3, the LCO would be exited prior to MODE 5, and MODE 5 entry is not a requirement.	No	Yes	No	No
02-28 LG	This change moves DCPD information inserted by LA 114/115 on containment spray and safety injection coincidence to the Bases.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-29 M	A new functional unit 9 is added to the DCPD CTS, per a License Amendment Request, that incorporates ACTION 20.1 (new) and the appropriate Surveillances for the residual heat removal (RHR) pump trip from low refueling water storage tank (RWST) level.	Yes	No	No	No
02-30 A	This change deletes Note (3) from Functional Unit 2.b. The slave relays listed in Note (3) were never associated with containment spray.	No, not in CTS.	No, not in CTS.	Yes	Yes
02-31 A	The *** Note for Functional Unit 6.g of CTS Table 3.3-3 is deleted. This note is no longer needed given that the ITS provides separate Condition entry on a per pump basis, as evaluated under CN 2-46-LS42, and given the adoption of ITS LCO 3.0.6 and the Safety Function Determination Program.	No, not in CTS.	No, not in CTS.	Yes	Yes
02-32 LS23	These changes affect the ACTION STATEMENTS for inoperable undervoltage or degraded voltage relays. For all trip functions, new statements are added which require that the diesel generator, which was made inoperable by the inoperable undervoltage or degraded voltage relay, may be declared inoperable in lieu of entering LCO 3.0.3. If the number of OPERABLE channels per bus is less than the Required Number, one hour is now allowed to restore the channel to operability before compensatory actions are required.	No, see CN-02-48-LS28.	Yes	No, see CN 2-18-LS-31.	No, see CN 2-18-LS-31.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-33 A	ACTION Statements 27a and 34a of CTS Table 3.3-3 are revised to delete information regarding the loss of one MSFIS channel since no restoration activity is required. See also CN 1-43-A. CTS Table 3.3-3 is revised to correct the implication that the MSFIS PLCs are associated with, or a part of, the SSPS. System design is described in the ITS 3.3.2 Bases. CTS Table 4.3-2 is revised to show that the quarterly SR is actually an Actuation Logic Test of the MSFIS PLC logic, initiated from the SSPS slave relays.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
02-34 LS34	ACTION Statement 19 of CTS Table 3.3-3 Functional Units 8a and 8b is revised to require an inoperable loss of power channel to be placed in trip within 6 hours and an allowance to be bypassed for surveillance testing for up to 4 hours is added rather than the current 1 hour and 2 hours, respectively. This is consistent with ISTS 3.3.5 Condition A.	No, maintained CTS.	No, can not bypass these channels.	Yes	Yes
02-35 A	Note (2) of DCPD CTS Table 4.3.2 is revised. The testing frequency was relaxed from monthly (M) to quarterly (Q) via License Amendment 102/101 but the note was not revised.	Yes	No	No	No
02-36 M	This change revises the APPLICABILITY of Functional Unit 7 to require operability when the associated DG is required to be OPERABLE by LCO 3.8.2 of the ITS.	Yes	No, see CN 2-32-LS-23.	No, see CN 2-18-LS-31.	No, see CN 2-18-LS-31.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-37 LG	CTS 3.3.2 Functional Unit 6.a requirement for manual initiation of an Auxiliary Feedwater pump is being deleted. The CTS Action Statement 24 with one channel inoperable requires declaring the applicable feedwater pump inoperable and entering the ACTIONS required by CTS 3.7.1.2. Therefore, this is consistent with the NUREG-1431 ITS 3.3.2 which does not have this Function identified and ITS 3.7.5 which has the ACTIONS consistent with those in the CTS ACTION 24.	No, retained CTS.	No, not in CTS.	Yes	No, retained in ITS.
02-38 LS35	For DCPD, delete MODE 4 Applicability from the Manual Initiation of MSIVs since the valves are not required to be OPERABLE in MODE 4 per CTS 3.7.1.5 or ITS 3.7.2.	Yes	No	No	No
02-39 LG	Move valve numbers in CTS ACTION 18 dealing with containment ventilation isolation to the Bases.	Yes	No, not in CTS.	No, not in CTS.	No, not in CTS.
02-40 A	This administrative change affects the manner in which the CTS 4.0.4 exception for testing the TDAFW pump is presented. The exception allows entry into MODE 3 to perform the TDAFW pump response time testing. In NUREG-1431, the CTS 4.0.4 exception from [CTS 3.7.1.2 has been interpreted so that it allows response time testing to be deferred as] is reflected in the ITS SR 3.3.2.10 NOTE.	Yes	Yes	No, already in CTS	No, already in CTS
02-41 LS36	ACTIONS Statements 27a and 34a of CTS Table 3.3-3 for the Automatic Actuation Logic and Actuation Relays (MSFIS), Functional Units 4.b.2 and 5.a.2, are revised to specify action "with the number of channels less than the required channels". Those ACTION Statements currently require action "with the number of OPERABLE channels <u>one</u> less than the minimum number of channels", but no action exists if none of the required channels per train are OPERABLE.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-42 LS38	This change revises the Applicability and ACTION Statement 37 for Functional Unit 6.d.1).c) of CTS Table 3.3-3 to reflect MODES 1 and 2 only. There is no Vessel ΔT in MODE 3 and no OPERABILITY requirements should be imposed. The maximum TTD should always be enabled in MODE 3.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
02-43 LG	Callaway's Operability requirements associated with the shutdown portion of one sequencer required to be OPERABLE in MODES 5 and 6, and which corresponds to the required DG, are moved to ITS 3.8.2 with the explicit link found in the ITS 3.8.2 LCO Bases.	No	No	No	Yes
02-44 A	Current Action Statement 32 of TS Table 3.3-3 for Functional Unit 32, RWST Level - Low-Low Coincident with Safety Injection, is revised to delete the last paragraph that states a channel may be tripped for up to 4 hours for surveillance testing to reflect the practical application of this allowance.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
02-45 LG	CTS Table 4.3-2 Note (3) is revised to move the "during refueling" clarification of the 18 month surveillance interval for the seven slave relays to the Bases for ITS SR 3.3.2.6, SR 3.3.6.5, and SR 3.3.7.5.	No, not in CTS.	No, not in CTS.	Yes	Yes
02-46 LS42	Relaxations associated with separate Condition entry for the RCP-Undervoltage, RCP-Underfrequency, and Auxiliary Feedwater - Trip of all Main Feedwater Pumps Functions are justified.	No, not in CTS.	No, already in CTS.	Yes	Yes
02-47 M	This change adds the Automatic Actuation Logic and Actuation Relays Function to the ESFAS Functional Unit for [Control Room Emergency Recirculation].	No, see CN 3-08-M.	Yes	No, already in CTS.	No, already in CTS.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-48 LS28	A new Action 15 is added and applied to function 7.a.2) and 7.b.1) that allows the affected Emergency Diesel Generator to be declared inoperable and requires entry into Specification 3.8.1.1 when more than one relay per bus is inoperable. Current ACTION 16 does not address the above situation and requires entry into LCO 3.0.3. This change is consistent with NUREG-1431.	Yes	No, see CN 2-23-LS23.	No, see CN 2-18-LS31.	No, see CN 2-18-LS31.
03-01 A	The requirements of this specification [CTS 3.3.3.1] are moved to [four] separate specifications in the improved TS. The RCS Leakage Detection requirements are moved to improved TS 3.4.15. [The Fuel Building requirements are moved to improved TS 3.3.8.] The Control Room requirements are moved to improved TS 3.3.7. [The Containment Ventilation Isolation is moved to improved TS 3.3.6.]	Yes	Yes	Yes	Yes
03-02 M	The requirements stipulated in ACTION [a] are moved to ITS Tables [3.3.6-1, 3.3.7-1 and 3.3.8-1], with explicit direction contained in the ITS ACTIONS Bases. The 4 hour AOT for setpoint adjustment is eliminated.	Yes	Yes	Yes	Yes
03-03 LG	The requirements associated with the criticality monitors are moved to a licensee controlled document. These monitors are required by 10CFR70.24; however, there is no requirement for [them] to be in the Technical Specifications [as criticality monitors. They are retained, however, as initiators of the Iodine Removal mode of the FHBVS for a fuel handling accident until RM-45A/B are installed].	Yes, moved to FSAR.	No, not in CTS.	Yes, moved to USAR Section 16.3.	Yes, moved to FSAR Section 16.3.
03-04 M	This change adds the Applicability for movement of irradiated fuel assemblies. The CTS Applicability of "All" MODES does not cover the movement of irradiated fuel assemblies when the core is offloaded.	Yes	Yes	No, see CN 3-12-A.	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-05 LS14	ACTION Statement [34] for the [Control Room Air Intake] [] radiation monitors have extended Completion Times, from [1 hour] to 7 days for one required channel inoperable.	Yes	Yes	Yes	Yes
03-06 A	ACTION c of CTS LCO 3.3.3.1 is revised to state the Specification 3.0.3 exception is [retained only for the Fuel Handling Building Radioactivity Instrumentation].	Yes	Yes	Yes	Yes
03-07 LS16	The Applicability for the Fuel [Handling] Building Exhaust radiation monitors has been revised to read "during movement of irradiated fuel assemblies in the fuel [handling] building." [The REQUIRED CHANNELS for Instrument 1.b. has been revised from one as specified by the CTS to two as specified by NUREG-1431 to provide single failure protection.]	Yes	No, not in CTS.	Yes	Yes
03-08 M	The DCPD CTS have been revised to include manual initiation of the fuel handling building and automatic initiation of the control room pressurization system. These systems are not classified as ESF functions in the CTS. This revision incorporates the Actuation Logic, Master Relay, and Slave Relay Tests included in NUREG-1431 for the CRVS and the TADOT for the manual actuation of both systems.	Yes	No	No	No
03-09 LS24	The CPSES Surveillance frequency for the performance of a CHANNEL OPERABILITY TEST for the radiation monitoring instrumentation channels would be extended from once per 31 days to once per 92 days. This change is consistent with the ITS.	No	Yes	No	No
03-10 LG	The DCPD descriptive information related to the Required Channels per normal intake is moved to the Bases.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-11 M	The CPSES ACTION STATEMENT for an inoperable gaseous radioactivity monitor is modified to require that the inoperable channel be restored within 4 hours. If not restored, continued operation is permitted (similar to the CTS allowance) provided the containment vent isolation valves are closed. The change is more restrictive or administrative in nature in that no time limits for the completion of any actions are specified in the CTS.	No	Yes	No	No
03-12 A	Wolf Creek's use of the Applicability "All" includes MODES 1, 2, 3, 4, 5, 6, and movement of irradiated fuel assemblies.	No, see CN 3-04-M.	No, see CN 3-04-M.	Yes	No, see CN 3-04-M.
03-13 M	The proposed change would provide an ACTION STATEMENT for the case where the number of Control Room air intake radiation level instrumentation channels was two less than required number of channels. The ACTION STATEMENT would require that the air supply from the affected air intake be secured or that the emergency recirculation mode be initiated and maintained. Otherwise, restore one channel within one hour or shutdown to MODE 3 in 6 hours and MODE 5 in the next 30 hours.	No, see CN 3-15-M.	Yes	No, already in CTS.	No, already in CTS.
03-14 LS29	This proposed change adds an ACTION and an allowed outage time of 4 hours for one inoperable Containment Ventilation Radiation instrumentation or actuation channel to the DCPD CTS.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-15 M	This change revises DCPD CTS ACTION 34 and adds new ACTION 36 to require appropriate MODE changes or condition changes for the CRVS with one or two inoperable normal intake monitors. These actions specify a shutdown requirement for MODES 1-4 that is one hour less than LCO 3.0.3, and immediate action for inoperability in MODE 5 or 6, and immediate action for inoperability during fuel movement. Refer also to CN 03-08-M, CN 03-04-M, and CN 3.3-51.	Yes	No	No	No
03-16 A	ITS 3.3.6 for DCPD includes MODES 1-4 and during movement of irradiated fuel assemblies within containment, in addition to MODE 6, in the LCO Applicability. These requirements are inferred in CTS 3.6.3 and are repeated here for clarity.	Yes	No	No	No
03-17 A	The CPSES restrictions on opening of the containment pressure relief valves is moved from the Radiation Monitoring Instrumentation specification in the CTS to ITS 3.6.3 and the ITS Administrative Controls Section 5.5.1 for the ODCM.	No	Yes	No	No
04-01 R	DCPD LCO 3.3.3.2, Movable Incore Detectors, is relocated to a licensee controlled document.	Yes, see Attachment 21, page 11.	No	No	No
05-01 R	DCPD LCO 3.3.3.3, Seismic Instrumentation, is relocated to a licensee controlled document.	Yes, see LAR 95-07	No	No	No
06-01 R	DCPD LCO 3.3.3.4, Meteorological Instrumentation, is relocated to a licensee controlled document.	Yes, see Attachment 21, page 13.	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
07-01 A	The requirements of CTS Table [3.3-9] are redefined on a functional basis with Required Channels. The point at which ACTION Statements are entered is unchanged.	No, already in CTS.	Yes	Yes	Yes
07-02 M	The shutdown requirement for inoperable Remote Shutdown controls is changed from HOT STANDBY to HOT SHUTDOWN.	No, already in CTS.	Yes	No, already in CTS.	No, already in CTS.
07-03	Not used.	N/A	N/A	N/A	N/A
07-04 LS15	This change extends the Remote Shutdown AOT from 7 days to 30 days.	No, already in CTS.	Yes	Yes	Yes
07-05 A	Consistent with the ITS, the modifications would clarify the requirement to be in HOT SHUTDOWN in 12 hours by replacing the requirement with a new requirement to be in HOT STANDBY in 6 hours and in HOT SHUTDOWN in the next 6 hours.	Yes	Yes	No, already in CTS.	No, already in CTS.
07-06 LG	The Readout Location and Total No. of Channels columns in CTS Table [3.3-9] have been moved to the Bases of improved TS 3.3.4. [Descriptive information related to the controls is also moved to the Bases.]	Yes	Yes	Yes	Yes
07-07	Not used.	N/A	N/A	N/A	N/A
07-08 TR2	The CPSES requirement to submit a special report if the number of remote shutdown monitoring instruments is less than the required number would be deleted from the CTS. This requirement is covered by other regulatory requirements.	No	Yes	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
07-09 LS43	Clarification is provided that Channel Checks are only required for normally energized instrumentation channels by adding "for each required instrumentation channel that is normally energized", per ITS [SR 3.3.3.1 and SR 3.3.4.1] to CTS [4.3.3.5.1 and 4.3.3.6].	No, retained CTS.	Yes	No, retained CTS.	Yes
07-10 LS26	In the CTS, the determination that the steam generators are available for decay heat removal, using instrumentation available at the remote shutdown panel, is based on steam generator level <u>and</u> auxiliary feedwater flow to the steam generator. In the ITS, it is recognized that either of these indications is sufficient.	Yes	Yes	No, not in CTS.	No, not in CTS.
07-11 A	CTS SR 4.3.3.5.3 is deleted. This is duplicated in CTS 3.7.1.2 and covered in ITS 3.7.5 for AFW System operability requirements.	No, not in CTS.	No, not in CTS.	Yes	No, retained in ITS.
07-12 A	New Note excludes neutron detectors from CHANNEL CALIBRATION consistent with CTS Table 4.3-1, Functional Unit 6, Note 4 and with improved TS SR 3.3.4.3.	No, not in CTS.	No, already in CTS.	Yes	Yes
08-01 A	This change, consistent with NUREG-1431 Rev. 1, revises "Channel" and "Instrument" to "Function."	Yes	Yes	Yes	Yes
08-02	Not used.	N/A	N/A	N/A	N/A
08-03 A	This change revises CTS Table [3.3-10] to clarify the number of channels required to be Operable. This is an administrative change which deletes the "Minimum Channels Operable" column []. The required actions are now based on one channel inoperable or two channels inoperable, rather than "less than the Total Number" or "less than Minimum Number."	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
08-04 LS17	Consistent with NUREG-1431 Rev. 1 (ITS 3.3.3 Required Actions C.1, E.1, and G.1), this change deletes the requirement to initiate an alternate means of monitoring within 72 hours when two channels of Containment Radiation Level [, main steam line radiation monitors or the plant vent radiation high range] are inoperable as specified in C[TS 3.3.3.6 ACTION d].	Yes	Yes	Yes	Yes
08-05 A	In conjunction with CN 8-03-A, this change rearranges the ACTION Statements for the single channel Functions (i.e., SG Water Level - Wide Range and AFW Flow Rate). No change in AOT is requested, therefore this is an administrative change.	No, see CN 8-11-LS-30.	No, not in CTS.	Yes	Yes
08-06 LG	Specific equipment ID numbers are moved to the ITS Bases.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
08-07 A	This CPSES-specific change revises the definition of Core Exit Temperature thermocouple channels. The change clarifies that 2 Core Exit Temperature thermocouple channels per quadrant per train with each channel consisting of 1 thermocouple, is the same as 2 channels per quadrant with each channel consisting of 2 thermocouples.	No	Yes	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
08-08 LS27	<p>Consistent with NUREG-1431 Rev. 1, this change extends the requirement to restore the inoperable channel to operable status from 7 days to 30 days. Consistent with NUREG-1431, this change extends the requirement to restore the inoperable channel to operable status from 48 hours to 7 days. Finally, consistent with NUREG-1431, this change replaces the requirement to shut down the plant with a requirement to submit a report describing an alternate, preplanned method of monitoring the process.</p> <p>Because the T_{hot} and T_{cold} channels are single channel functions (1/loop) with adequate alternate functions available, the action statements for these functions are revised consistent with the philosophy of NUREG-1431. The restoration time for inoperable channels with adequate alternate functions is decreased from 90 days in CTS to 30 days in the improved TS; however, the option of providing a written report describing an alternate, preplanned method of monitoring the process is allowed. If the above conditions are not met, 7 days are allowed for function restoration prior to initiating a required shutdown.</p> <p>ACTION 33 has been deleted because, after revision to be consistent with NUREG-1431, it is identical to ACTION 32.</p> <p>These changes are less restrictive in that the required channel restoration times are increased and the option of submitting special reports in lieu of a plant shutdown is provided.</p>	No	Yes	No	No
08-09	Not used.	N/A	N/A	N/A	N/A

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
08-10	Not used.	N/A	N/A	N/A	N/A
08-11 LS30	This change revises the DCPD CTS 3.3.3.6 to conform to NUREG-1431 Revision 1 and revises CTS Table 3.3-10 to both add and delete instruments per the Reviewer's Note on ISTS Table 3.3.3-1.	Yes	No	No	No
09-01 LG	The explosive gas monitoring instrumentation will be controlled by the Explosive Gas Monitoring Program established in accordance with ITS 5.5.12.	Yes, see Attachment 21, page 15.	Yes	No, already moved to Administrative Controls section (OL Amendment No. 89).	No, already moved to Administrative Controls section (OL Amendment No. 103).
10-01 R	The Turbine Overspeed Protection System is relocated to a licensee controlled document.	Yes, see LAR 95-07.	Yes, relocated to TRM.	No, already relocated to USAR (OL Amendment No. 89).	No, already relocated to FSAR Section 16.3 (OL Amendment No. 103).
11-01 R	LCO 3.3.3.7, Chlorine Detection Systems, is relocated to a licensee controlled document.	Yes, see LAR 95-07.	No, not in CTS.	No, not in CTS.	No, not in CTS.

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)
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	LS-14.....	37
	LS-15.....	Not applicable
	LS-16.....	39
	LS-17.....	41
	LS-18.....	43
	LS-19.....	Not applicable
	LS-20.....	Not used
	LS-21.....	Not applicable
	LS-22.....	Not applicable
	LS-23.....	Not applicable
	LS-24.....	Not applicable
	LS-25.....	Not used
	LS-26.....	45
	LS-27.....	Not applicable
	LS-28.....	47
	LS-29.....	49
	LS-30.....	51
	LS-31.....	Not applicable
	LS-32.....	Not applicable
	LS-33.....	Not applicable
	LS-34.....	Not applicable
	LS-35.....	56
	LS-36.....	Not applicable
	LS-37.....	Not applicable
	LS-38.....	Not applicable
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NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)
CONTENTS

LS-41.....Not applicable
LS-42.....Not applicable
LS-43.....Not applicable

V. Recurring No Significant Hazards Considerations - "TR"

TR-1.....Not applicable
TR-2.....Not applicable

I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3...." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"

10CFR50.92 EVALUATION

FOR

ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"

(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION
BASES, FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"

(Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

In CTS SR [4.3.1.2 and 4.3.2.2], the active verb is changed from "demonstrated" to "verified." This allows Reactor Trip System and ESFAS sensor response time verifications to be performed per WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Test Requirements." This change is consistent with Traveler TSTF-111, Rev. 1 which revises the Bases for ITS SR 3.3.1.16 and SR 3.3.2.10 to allow the elimination of pressure sensor response time testing.

In 1975 response time testing (RTT) requirements were included in the Westinghouse Standard Technical Specifications and were required for all plants licensed after that date.

The CTS contain definitions for both Reactor Trip System and Engineered Safety Feature (ESF) response times. The response time definitions are:

"The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage."

"The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable."

The current {method requires} that the response time may be measured by any series of sequential, overlapping, or total channel test measurements such that the total channel response time is measured. This approach is also consistent with ISA Standard 67.06. Given this guidance and the complexity of testing an entire instrument channel from the sensor to the final device, plant surveillance procedures test the channels in several steps. One individual step is the instrument sensor. Separate procedures using specialized test equipment are used solely for testing the sensors.

The purpose of this evaluation is to determine if the deletion of periodic response time testing could be justified for specific pressure, level, and flow functions that utilize pressure and differential pressure sensors. IEEE Standard 338-1977 defines a basis for eliminating RTT. Section 6.3.4 states:

"Response time testing of all safety-related equipment, per se, is not required if, in lieu of response time testing, the response time of the safety system equipment is verified by functional testing, calibration check, or other tests, or both."

WCAP-13632-P-A Rev. 2 provides the technical justification for deletion of periodic response time testing of selected pressure sensing instruments. The program described in the WCAP utilizes the methods contained in EPRI Report NP-7243 Rev. 1, "Investigation of Response Time Testing Requirements," for justifying elimination of response time testing surveillance requirements on certain pressure and differential pressure sensors. The EPRI report justifies the elimination of response time testing based on Failure Modes and Effects Analysis (FMEA) that show that component degradation that impacts pressure sensor response time can be detected in other routine tests such as calibration tests. The report concludes that sensor RTT is redundant to other technical specification surveillance requirements such as sensor calibrations. The EPRI report only applies to those specific sensors included in the FMEA.

To address other sensors installed in Westinghouse designed plants, the WCAP contains a similarity analysis to sensors in the EPRI report or an FMEA to provide justification for elimination of response time

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (continued)

testing requirements for those other sensors. The specific sensors installed at [Callaway] and that require RTT are listed below:

- Steam Generator Water Level	Rosemount/Model 1154
- Pressurizer Pressure	Rosemount/Model 1154
- Steamline Pressure	Barton/Model 763 and Rosemount/Model 1154
- Containment Pressure	Rosemount/Model 1154
- Reactor Coolant Flow	Rosemount/Model 1154

The basis for eliminating periodic response time testing for each sensor is discussed in the WCAP and/or the EPRI report. These reports provide justification that any sensor failure that significantly degrades response time will be detectable during surveillance testing such as calibration and channel checks.

The applicability of the generic analysis of WCAP-13632-P-A Revision 2 has been confirmed for [DCPP]. Each of the above transmitters is included in Table 9-1 of WCAP-13632. In addition, the following discussion addresses the four actions raised in the NRC SER dated September 5, 1995:

- (a) A hydraulic response time test will be performed on any new or refurbished transmitter, prior to declaring the affected channel operable, to determine an initial sensor-specific response time value.
- (b) A hydraulic response time test will be performed on units that use capillary tubes after initial installation of replacement transmitters or following any maintenance or modification activity that could damage the capillary tubing or degrade the response time characteristics of installed sensors.
- (c) [DCPP] does not utilize pressure sensors that incorporate a variable damping feature in the RTS or ESFAS channels that are required to have their response times verified.
- (d) {DCPP has established an enhanced monitoring program for those Rosemount transmitters that require monitoring per NRC Bulletin 90-01 and Supplement 1. Some of those transmitters are RTS or ESFAS transmitters that require RTT. These transmitters will remain in the enhanced monitoring program until they are replaced or the alternative method of performing periodic drift monitoring as described below is initiated:
 1. Ensure that operators and technicians are aware of the Rosemount transmitter loss of fill-oil issue and make provisions to ensure that technicians monitor for sensor response time degradation during the performance of calibrations and functional tests of these transmitters, and
 2. Review and revise surveillance testing procedures, if necessary to ensure that calibrations are being performed using equipment designed to provide a step function or fast ramp in the process variable and that allows simultaneous monitoring of both the input and output response of the transmitter under test. Thus allowing, with reasonable assurance, the recognition of significant response time degradation.}

Based on these results, the Technical Specifications are revised to indicate that the system response time shall be verified utilizing a sensor response time justified by the methodology described in WCAP-13632-P-A Revision 2. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in-place, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (continued)

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The same RTS and ESFAS instrumentation is being used. The time response allocations/modeling assumptions in the Chapter 15 analyses are still the same, only the method of verifying time response is changed. The proposed change will not modify any system interface and could not increase the likelihood of an accident since these events are independent of this change. The proposed activity will not change, degrade or prevent actions or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the FSAR. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change will not alter the normal method of plant operation. No transmitter performance requirements will be affected. This change does not alter the performance of the pressure and differential pressure transmitters used in the plant protection systems. All sensors will still have response time verified by test before placing the sensor in operational service and after any maintenance that could affect response time. Changing the method of periodically verifying instrument response for certain sensors (assuring equipment operability) from time response testing to calibration and channel checks will not create any new accident initiators or scenarios. Periodic surveillance of these instruments will detect significant degradation in the sensor response characteristic. No new transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (continued)

proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. This change does not affect the total system response time assumed in the safety analysis. The periodic system response time verification method for selected pressure and differential pressure sensors is modified to allow use of actual test data or engineering data. The method of verification still provides assurance that the total system response is within that defined in the safety analysis, since calibration tests will detect any degradation which might significantly affect sensor response time. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS1" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A new ACTION Statement [2.1] is created which is essentially the same as current ACTION Statement [6], and is similar to current ACTION Statement [2], but does not require a reduction in THERMAL POWER to less than 75% RTP or the measurement of the QPTR if above 75% RTP. This new ACTION Statement is applied to the Power Range Neutron Flux, High Positive Rate [, High Negative Rate] trip functions. Since these are rate functions, their effectiveness is not improved by reducing power.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards, as applied to the relaxation applicable to the power range neutron flux, high positive rate [and high negative rate] trip functions:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the ACTION associated with an inoperable channel in the power range neutron flux, high positive rate [and high negative rate] trip function. No power reduction below 75% RTP or QPTR monitoring above 75% RTP would be required since these actions have no basis for this rate trip function.

The high positive rate [and high negative rate] trip functions are insensitive to the static power level. The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. [] The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in required ACTION will not affect the normal method of plant operation and, in fact, will relieve the operators from performing actions not required for plant safety. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS2" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

With one intermediate range neutron flux channel inoperable, current ACTION Statement [3.a] applies below the P-6 interlock. For those times that the plant is above P-6 but below 10% RTP (the P-10 interlock setpoint), current ACTION Statement [3.b] applies. ACTION Statement [3.b] is revised to establish a 24 hour Completion Time for channel restoration or changing the power level to either below P-6 or above P-10. The intermediate range neutron flux channels provide protection between these power levels and the APPLICABLE MODES have been revised accordingly. The source range neutron flux detectors provide protection below P-6 and the power range neutron flux detectors provide protection above P-10. The addition of the 24 hour Completion Time (current TS has no Completion Time) limits the window of operation during which the intermediate range neutron flux trip function provides protection in a 1 of 1 logic configuration and ensures the low probability of occurrence of a reactivity transient occurring during this time period that would require an intermediate range flux trip. Although this change is less restrictive since a power increase is an allowed option, the ACTION Statement would ensure protection by entering the range of the four power range neutron flux channels.

With both intermediate range neutron flux channels inoperable in MODE 1 (below P-10) and MODE 2 (above P-6), LCO 3.0.3 would be entered under the current TS and the plant would have to be in MODE 3 within 7 hours. With both intermediate channels inoperable, new ACTION Statement [3.1] requires immediate suspension of operations involving positive reactivity additions and a power reduction below P-6 within 2 hours. New ACTION Statement [3.1] is less restrictive since a reduction to MODE 3 would no longer be required; however, the current TS are overly conservative in this area. Below P-6 the source range channels provide protection; therefore, the required action for both intermediate range channels inoperable should be to exit plant conditions where this trip function provides protection. New ACTION Statement [3.1] will preclude any power level increase and require a controlled power reduction to less than P-6 where the source range channels provide protection. The 2 hour Completion Time ensures the low probability of occurrence of an event during this period that may require the protection afforded by the intermediate range neutron flux trip. These actions actually provide a more timely and appropriate redress to the condition than entering LCO 3.0.3.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards, as applied to the relaxations applicable to the intermediate range neutron flux trip function:

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3 (continued)

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change relaxes the ACTIONS associated with inoperable intermediate range neutron flux channel(s). With one or both intermediate range neutron flux channels inoperable above the P-6 interlock but below 10% RTP, timely actions would be taken to exit plant conditions that may rely on this trip function for protection. The proposed change in the ACTION Statements will not affect any of the analysis assumptions for any of the accidents previously evaluated. No FSAR Chapter 15 analyses specifically credit this trip function. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in required ACTIONS will not impact the normal method of plant operation; however, operational flexibility will be added by allowing a power increase as a corrective measure if one intermediate range neutron flux channel is inoperable. The option of increasing power is consistent with the philosophy of placing the plant in a condition where protection is assured, in this case by the power range neutron flux channels. No new accidents are created since rod withdrawal events are analyzed with the initial power level at full power as well as subcritical; or low power startup conditions, i.e. the change in power level does not create a new kind of accident. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS3" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The APPLICABILITY for the Reactor Trip on Turbine Trip function is modified such that this function is only required to be OPERABLE above the P-9 interlock setpoint (15% RTP). This is acceptable since the trip function is blocked below P-9. CTS ACTION Statement 7 is revised accordingly and new note (j) is applied to Functional Units 17.a. and b. for the trip functions associated with Low Autostop Oil Pressure and Turbine Stop Valve Closure. The ACTION is also revised to allow bypassing a tripped Low Autostop Oil Pressure channel for up to four hours for surveillance testing other channels.

Current ACTION Statement 7 allows continued operation with one inoperable channel as long as it is placed in trip within 6 hours. There is no associated action if an inoperable channel is not placed in trip within 6 hours nor is there an action for multiple inoperable channels; therefore, LCO 3.0.3 would be invoked. Revised ACTION Statement 7 allows continued operation with one or more inoperable channels as long as they are placed in trip within 6 hours or THERMAL POWER is reduced below P-9 within 10 hours, thus LCO 3.0.3 would no longer be applicable.

Revised ACTION Statement 7 is consistent with the ITS philosophy of reducing power to enter a condition where a function is not required to be OPERABLE. This is less restrictive than the CTS which would require entry into LCO 3.0.3. Entry into LCO 3.0.3 is overly conservative in the above situations since this trip function provides an anticipatory reactor trip function only above P-9.

A note is added per NUREG-1431 to allow the Low Auto Stop Oil Pressure trip to be bypassed for up to four hours. There is no need for the 4 hour bypass allowance for surveillance testing of the Turbine Stop Valve Closure, since all 4 channels are required to trip the turbine; however, the bypass allowance is desirable when 2 channels satisfy the trip logic to reduce the possibility of a spurious trip during testing. Bypassing one channel for testing changes the trip coincidence to a two out of two thus maintaining its operability with only a slight reduction in reliability. In addition, the potential for a turbine trip during the bypass period for other than a Low Autostop Oil Pressure trip is small. The reduction in reliability is acceptable since the potential for a transient due to a reactor trip is reduced by allowing the bypass.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4 (continued)

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the ACTIONS associated with inoperable channel(s) in the Reactor Trip on Turbine Trip function by keeping the end point of the shutdown action above the CTS requirement if inoperable channel(s) are not placed in trip within 6 hours or if multiple Low Autostop Oil Pressure channels are inoperable. The new ACTION Statement would reduce power to less than P-9 (50% RTP) in these situations as compared to entry into LCO 3.0.3 (power \leq 5% RTP) in the current TS. The proposed change in the ACTION statements will not affect any of the analysis assumptions for any of the accidents previously evaluated. This trip function is anticipatory only and is not credited in any FSAR Chapter 15 accident analyses. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in ACTIONS will not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS4" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, an NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

[ACTION Statement [10] is revised to note that the 2 hour [train and] reactor trip breaker bypass allowance for [train or] breaker surveillance testing can also be used for maintenance. This change does not impact the conclusions of WCAP-10271-P-A, Supplement 2, Rev. 1 since there is no change to the bypass time.] ACTION Statement [9] is [also] revised to require restoration of an inoperable Reactor Trip Breaker (RTB) within 1 hour or the plant must be in HOT STANDBY within the next 6 hours. This is less restrictive than the current TS since an additional hour is provided for the transition to MODE 3.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. As noted in the Bases of NUREG-1431 Rev. 1, the Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by current LCO 3.0.3 for shutdown actions. The proposed shutdown requirement Completion Time change would result in an additional hour to achieve MODE 3. By allowing a shutdown time based on operating experience, this change would reduce the chances of an operator error or challenge to plant systems that could result from the more restrictive requirement in the current TS. The probability that an accident would occur during the 1 hour extension allowed by the proposed change is extremely small. [Use of the 2 hour [train or] reactor trip breaker bypass allowance for maintenance will not change the unavailabilities used to determine core damage frequencies in WCAP-10271-P-A, Supplement 2, Rev. 1 since the bypass time allowance is not extended.] The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change will not affect the normal method of plant operation. Only the duration of operation in the action statement is affected. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS6" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

In current ACTION Statement [2c], the requirement to reduce the power range neutron flux high trip setpoint is deleted. The loss of one power range neutron flux channel does not impact the reliability of the reactor trip system because the channel is required to be placed in trip. With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. To address the impact on measuring QPTR, the QPTR TS provides conditional notes to the surveillances that verify QPTR. Although the loss of one power range channel may affect the ability to measure QPTR, there is no basis for reducing the power range high flux trip setpoint and increasing the potential for an inadvertent reactor trip. The existing surveillance requirements in the QPTR TS provide adequate remedial measures (increased surveillance frequency and/or different method for monitoring QPTR) when an inoperable power range channel affects the input to QPTR.

In addition, monitoring of QPTR every 12 hours will no longer be required if the input to the QPTR from the power range neutron flux channels remains OPERABLE. A consistent completion time of 12 hours is imposed on either of the three required actions, i.e. power derate, QPTR surveillance, or shutdown to MODE 3.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the ACTION associated with an inoperable power range neutron flux channel. The inoperable channel is placed in the tripped condition, therefore the reliability of the RTS is unaffected. A consistent completion time of 12 hours is imposed on either of the three required actions, i.e. power derate, QPTR surveillance, or shutdown to MODE 3. While this represents a relaxation for the power derate completion time, currently 4 hours, this time is not an assumption in any accident analysis. For the purposes of the FSAR Chapter 15 accident analyses, placing the inoperable channel in trip within 6 hours, which is unchanged, satisfies the analysis assumptions on RTS availability. This relaxation is contrasted by the more restrictive change that

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7 (continued)

if the derate or QPTR surveillance can not be performed, the plant must be shutdown to MODE 3. Currently, failure to meet ACTION Statement 2 would place the plant in LCO 3.0.3, with 7 more hours to get to MODE 3. The proposed change in the ACTION Statement will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in the ACTION Statement will not affect the normal method of plant operation, but will reduce the possibility of an unplanned plant transient that might occur during the adjustment of the high flux trip setpoint. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS7" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change reflects a revision to current ACTION Statement [6]. If the requirements of current ACTION Statement [6] are not met, LCO 3.0.3 would be entered. This ACTION Statement is revised to state that if the ACTION requirements are not met, THERMAL POWER must be reduced to below the P-7 interlock setpoint within the next 6 hours. Most of the Functional Units that impose ACTION Statement [6], Pressurizer Pressure - Low, Pressurizer Water Level - High, Reactor Coolant Flow - Low, Two Loops (above P-7 and below P-8), RCP Undervoltage, and RCP Underfrequency, are automatically blocked below P-7 and an Applicability Note has been added accordingly. The Reactor Coolant Flow - Low (Single Loop) reactor trip function does not have to be OPERABLE below the P-8 setpoint; however, the Required Action must take the plant below the P-7 setpoint, if an inoperable channel is not tripped within 6 hours, due to the shared components between this function and the Reactor Coolant Flow - Low (Two Loops) trip function.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the ACTION Statement associated with an inoperable channel in CTS Table 3.3-1 Functional Units [9, 11, 12.a, 12.b, 14, and 15] by keeping the end point of the shutdown action above the CTS requirement if an inoperable channel isn't placed in trip within 6 hours. The new ACTION Statement would reduce power to less than P-7 (10% RTP) within the next 6 hours in this situation as compared to entry into LCO 3.0.3 (power \leq 5% RTP) in the current TS. The proposed change in the ACTION Statement will not affect any of the analysis assumptions for any of the accidents previously evaluated. An LCO 3.0.3 shutdown to \leq 5% RTP is not required to meet the initial conditions of any accident analysis crediting these trip functions. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8 (continued)

change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in the ACTION Statement will not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS8" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change reflects a relaxation in the performance of COTs prior to startup. These COTs, for the Power Range Neutron Flux-Low, Intermediate Range Neutron Flux, and Source Range Neutron Flux trip functions, will not be required if performed within the previous 92 days. Note [(1)] of current TS Table 4.3-1 has been revised to extend this period from 31 days to 92 days. Since the COT is valid for 92 days at all other times, there should be no need to repeat it during startup if one has been performed within the prior quarter.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. A Channel Operational Test demonstrates channel operability every 92 days, consistent with the requirements of WCAP-10271 and its Supplement 1 for the RTS. Since it remains valid for that time period, there is no need to perform another COT during startup if one had been performed within the prior 92 days. This will not affect the operability or functionality of the NIS channels. Their calibration is still valid and daily channel checks are still performed. The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in the COT performance Note will

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9 (continued)

not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS9" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Surveillances on the Source Range Neutron Flux trip function are reorganized to reflect plant status in accordance with NUREG-1431 Rev. 1. New Note [(19)] requires that the quarterly COT be performed within 4 hours after reducing power below the respective source range instrumentation Applicabilities, if not performed within the previous [92] days. Since the COT is valid for [92] days, there is no need to repeat it if one has been performed within the prior [quarter]. The 4 hour allowance permits a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the Reactor Trip Breakers are opened and this trip function no longer provides protection. Since the current TS has no Specification 4.0.4 exception, this 4 hour allowance is less restrictive.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. This change allows a 4 hour deferral of a Channel Operational Test after entering the Applicability of the source range trip. This will not affect the operability or functionality of the source range channels. Their calibration is still valid and daily channel checks are still performed. The probability that an accident would occur during the 4 hour extension allowed by the proposed change is extremely small. The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. No FSAR Chapter 15 accident analyses credit this reactor trip function. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change will not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS10" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS11
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

[Note (a) is added to current TS Table 3.3-3 for the Steam Line Isolation Function to state that the LCO requirements are not applicable in MODES 2 and 3 when the MSIVs are closed and deactivated.]
Note [(b)] is added to current TS Table [3.3-3] for the Feedwater Isolation and Turbine Trip Function to state that the LCO requirements are not applicable when the [MFIVs, MFRVs and the associated bypass valves] are closed and deactivated [or isolated by a closed manual isolation valve]. When these valves are closed and deactivated [or isolated by a closed manual isolation valve], they are already performing their safety function.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the Applicability associated with the [Steam Line Isolation and] Feedwater Isolation ESFAS functions. Those functions are accomplished when the associated valves are closed and deactivated [or isolated by a closed manual isolation valve], whether that closure is as a result of automatic isolation circuitry or operator action. Operability requirements on actuation circuitry are not applicable if the valves are closed and deactivated [or isolated by a closed manual isolation valve]. The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS11 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in Applicability will not affect the normal method of plant operation as it recognizes the safety function is accomplished as long as the valves are closed and deactivated [or isolated by a closed manual isolation valve]. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS11" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

ACTION Statement[s] [30 and 38] in current TS Table [3.3-6] for the [Fuel Building Radioactivity monitors and Control Room Air Intake Gaseous Radioactivity monitors] have been revised to allow 7 days for restoration of an inoperable channel prior to isolation of the normal ventilation systems and initiation of the ESF HVAC systems. The current allowed outage time (AOT) for [both] ACTION Statement[s] is [72 hours]. The 7 day AOT for one channel inoperable is the same as that allowed by the ACTION Statement[s] of current TS LCO[s] 3.7.6 and 3.7.7] for one train of the mechanical portion of the [ESF] HVAC system inoperable.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the AOT associated with an inoperable radiation monitor channel in the ESF HVAC systems for the [fuel building and] control room. The proposed change in the AOT will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Mechanical portions of the systems can be inoperable, rendering one train inoperable, for up to 7 days. Therefore, this AOT change has effectively been approved by the precedent established in the plant systems TS. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in AOT will not impact the normal method of plant operation. More thorough troubleshooting and restoration activities are possible with an extended AOT. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no adverse effect on control room habitability conditions of offsite doses. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS14" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The Applicability for the [Fuel Building Gaseous Radioactivity] monitors in current TS Table 3.3-6 is revised to read "during movement of irradiated fuel assemblies in the fuel building." The current TS Applicability, at all times with irradiated fuel in the fuel storage areas or fuel [handling] building, is overly restrictive since the radiation monitors' function is to ensure automatic actuation of the [Fuel Handling Building Ventilation System to the iodine removal mode] when the potential for a fuel handling accident exists. That potential exists only when fuel assemblies are being moved. These monitors could also provide a mitigation function in the event the spent fuel pool were excessively drained; however, that scenario is not credible given the design provisions to ensure spent fuel pool integrity and the monitoring system provided to measure pool level and annunciate low level, as described in FSAR Sections [9.1.2, 9.1.3, and 9.3.3].

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the Applicability for the [Fuel Building Gaseous Radioactivity] monitors. The proposed change in the Applicability will not affect any of the analysis assumptions for any of the accidents previously evaluated. The only accident analysis crediting these radiation monitors is the Fuel Handling Accident in the fuel building. The Applicability ensures the monitors' availability for that accident by virtue of the operability requirements imposed any time irradiated fuel is moved in the fuel building. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in Applicability will not impact the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. Offsite doses in the event of a Fuel Handling Accident will not be affected. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS16" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS17
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change deletes the requirement to initiate the preplanned alternate method of monitoring containment radiation [or reactor vessel water level] within 72 hours when two channels are inoperable. This makes ACTION [c] of current TS LCO [3.3.3.6] the same as ACTION [b], except that a plant shutdown is not needed if the 7 day AOT is not met for these functions with preplanned alternates. Containment radiation indication is used to assess whether the fuel cladding and reactor coolant pressure boundaries have been breached such that a significant portion of the core activity inventory is available for release to the environs. The preplanned alternate for this variable uses the PASS system which is administratively controlled outside the TS. [The preplanned alternates for RVLIS include a combination of the core exit thermocouples, RCS wide range hot and cold leg temperature, wide range RCS pressure, pressurizer level, and RCS subcooling monitor indications to verify adequate core cooling.] These variables are already continuously monitored under the LCO and do not require an action directing that they be initiated.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change involves a relaxation regarding the deletion of the 72 hour initiation of the preplanned alternate method of monitoring containment radiation [or reactor vessel water level] if two channels are inoperable. The proposed change in the ACTION Statement will not affect any of the analysis assumptions for any of the accidents previously evaluated. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS17 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change will not impact the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS17" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS18
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The APPLICABILITY for the Reactor Trip on Reactor Coolant Pump Breaker Position above P-7 Trip function is modified such that this function is only required to be OPERABLE above the P-7 interlock setpoint (10% RTP). This is acceptable since the trip function is blocked below P-7. CTS ACTION Statement 9 is deleted and revised ACTION 6 and new note (g) is applied to Functional Unit 19 for the trip functions associated with Reactor Coolant Pump Breaker Position. The ACTION is also revised to allow bypassing a tripped Reactor Coolant Pump Breaker Position channel for up to four hours for surveillance testing other channels.

Current ACTION Statement 9 applies to the Reactor Coolant Pump Breaker Position trip function and allows continued operation with one inoperable channel as long as it is placed in trip within 6 hours. There is no associated action if an inoperable channel is not placed in trip within 6 hours nor is there an action for multiple inoperable channels; therefore, LCO 3.0.3 would be invoked. Revised ACTION Statement 6 allows continued operation with one or more inoperable channels as long as they are placed in trip within 6 hours or THERMAL POWER is reduced below P-7 within 12 hours, thus LCO 3.0.3 would no longer be applicable.

Revised ACTION Statement 6 is consistent with the ITS philosophy of reducing power to enter a condition where a function is not required to be OPERABLE. This is less restrictive than the CTS which would require entry into LCO 3.0.3. Entry into LCO 3.0.3 is overly conservative in the above situations since this trip function provides an anticipatory reactor trip function only above P-7.

A note is added per NUREG-1431 to allow the Reactor Coolant Pump Breaker Position trip to be bypassed for up to four hours. The bypass allowance is desirable when 2 channels satisfy the trip logic to reduce the possibility of a spurious trip during testing. Bypassing one channel for testing changes the trip coincidence to a two out of three versus two out of four thus maintaining trip operability with only a slight reduction in reliability. In addition, the potential for a reactor trip during the bypass period is small. The reduction in reliability is acceptable since the potential for a transient due to a reactor trip is reduced by allowing the bypass.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves a no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

LS18 (continued)

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the ACTIONS associated with an inoperable channel by keeping the end point of the shutdown action above the CTS requirement if the inoperable channel is not placed in trip within 6 hours. The new ACTION Statement would reduce power to less than P-7 (10% RTP) in these situations as compared to entry into LCO 3.0.3 (power \leq 5% RTP) in the current TS. The proposed change in the ACTION statements will not affect any of the analysis assumptions for any of the accidents previously evaluated. This trip function is anticipatory only and is not credited in any FSAR Chapter 15 accident analyses. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in ACTIONS will not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS18" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, an NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS26
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The Remote Shutdown System is required to provide equipment at appropriate locations outside the control room with a capability to place and maintain the unit in a safe condition in MODE 3 (prior to control room evacuation, the reactor would have been verified to be shutdown). Under this condition, decay heat would be removed from the RCS through the steam generators. Steam is released through the main steam safety and/or relief valves and steam generator inventory is maintained by the Auxiliary Feedwater System. Indication of either the steam generator level or the auxiliary feedwater flow rate is sufficient to ensure that the steam generator inventory is being replenished.

In the CTS, the determination that the steam generators are available for decay heat removal, using instrumentation available at the remote shutdown panel, is based on steam generator level and auxiliary feedwater flow to the steam generator. In the ITS, it is recognized that either of these indications is sufficient; therefore, the decay heat removal via steam generators function may be satisfied through the availability of either of these two functions.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change to require only one of two available indications of the capability to use the steam generators to remove decay heat from the RCS can not affect the probability of an accident. The consequences of the accidents considered in the safety analyses are largely independent of operator action; however, even with the reduced requirements for indication of steam generator heat removal capability, sufficient information is available to the reactor operators such that the assumed operator actions will not be affected. Therefore, neither the probability nor the consequences of an accident previously evaluated will be affected by this change.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

2. Does the change create the possibility of a new or different kind of accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change will not alter the normal method of plant operation. There will be no decrease in the information available to the reactor operator in the event of an accident that would lead the operator to take inappropriate actions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change affects neither the relevant event acceptance criteria for any analyzed event nor any assumed failure point. Therefore, there will be no effect on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS26" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10.CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS28
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change incorporates a new required ACTION for CTS 3.3.2 Table 3.3-3, Functional Unit 7.a.2) and 7.b.1). This new ACTION 15 would require that for either one or two relays per bus inoperable, that the associated DG be declared inoperable and that the ACTION Statements for TS 3.8.1.1 be met. The CTS do not have an ACTION for two inoperable relays, thus LCO 3.0.3 must be entered. License Amendment 97-02 which included ACTION 15 was submitted on February 27, 1997.

Two levels of undervoltage detection and automatic transfer are provided for the 4kV vital buses to start the DGs and to transfer vital loads to the DGs. The first level of undervoltage protection (FLUR) detects the loss of bus voltage and has sufficient time delay to allow the transfer of the vital buses to the startup transformer.

The second level of undervoltage protection (SLUR) is intended to protect against a degraded voltage condition, and has two sequential time delays for DG starting and loading.

The load shed FLUR and SLUR contacts are each connected in series for two-out-of-two logic. This assures that a single failure of a load shed FLUR or SLUR does not cause an unnecessary transfer of the vital buses to the DGs. Since both relays are required to be OPERABLE to cause an undervoltage actuation, the current ACTION 16 recognizes that the undervoltage function will not operate with one relay inoperable. Inoperability of the second relay does not change conditions recognized by the current ACTION Statement, thus the ACTION Statement should be the same for one or two relays inoperable.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the required actions in that LCO 3.0.3 is not required to be entered. However, the condition of two relays inoperable does not change the fact that with one relay inoperable, the undervoltage actuation will not actuate. The proposed change will not affect the probability of any accident initiators nor will it affect the ability of the safety-related equipment to perform its intended

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS28 (continued)

function. Therefore, neither the probability nor the consequences of an accident previously evaluated will be affected by this change.

2. Does the change create the possibility of a new or different kind of accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in ACTION will not alter the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change affects neither the relevant event acceptance criteria for any analyzed event nor any assumed failure point. Therefore, there will be no effect on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS28" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS29
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed change adds a new ACTION 37 which allows 4 hours for restoration of an inoperable train or channel if one of the required channels of the Containment Ventilation Isolation Radiation Monitor instrumentation is inoperable in accordance with NUREG-1431. The CTS via ACTIONS 18 and 33 requires that for one or two instrument or actuation channels inoperable that CTS LCO 3.6.3 or LCO 3.9.9 be entered as applicable. The revised specification will require that the ITS LCO 3.6.3 or 3.9.4 be entered if the instrument or actuation channel cannot be restored to an OPERABLE condition within the allowed outage time (AOT). Both of these specifications require that if the penetrations cannot be closed automatically that the penetrations be closed in the manner specified. The 4 hours allowed to restore the affected channel is justified based on the low probability of events requiring containment ventilation actuation during this time period, and recognition that the remaining channel is OPERABLE and would respond.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the required actions in that the AOT for one monitor or channel is permitted to be 4 hours. The proposed addition of the AOT will not affect the probability of any accident initiators nor will it affect the ability of the safety-related equipment to perform its intended function. Therefore, neither the probability nor the consequences of an accident previously evaluated will be affected by this change.

2. Does the change create the possibility of a new or different kind of accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in AOT will not alter the normal method of plant operation. More thorough trouble shooting and restoration activities are possible with this additional AOT. No new accident scenarios, transient precursors, failure mechanisms, or

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS29 (continued)

limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change affects neither the relevant event acceptance criteria for any analyzed event nor any assumed failure point. Therefore, there will be no effect on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS29" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS30
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change to CTS 3/4.3.3.6, "Accident Monitoring Instrumentation," revises the instrumentation used for monitoring plant conditions following an accident. The post accident monitoring instrumentation listed in TS Table 3.3-10 is revised to include all Diablo Canyon Power Plant (DCPP) Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Type A and Category 1 instrumentation.

The following DCPP RG 1.97 Type A and/or Category 1 accident monitoring instrumentation is added to TS Table 3.3-10:

- a. Containment Pressure (Wide Range)
- b. Steam Generator Water Level (Wide Range)
- c. Neutron Flux (Wide Range NIS)
- d. Containment Isolation Valve Position
- e. Containment Hydrogen Concentration
- f. Condensate Storage Tank Level

The following non-type A and non-Category 1 DCPP RG 1.97 instrumentation is deleted from TS Table 3.3-10:

- a. Reactor Coolant System (RCS) Subcooling Margin Monitor
- b. Power Operated Relief Valve (PORV) Position Indicator
- c. PORV Block Valve Position Indicator
- d. Safety Valve Position Indicator
- e. Main Steam Line Radiation Monitor
- f. Plant Vent Radiation Monitor-High Range

ACTION Statement a. is modified to allow an outage time of 30 days, versus the CTS 7 days, for one channel less than the required number of channels, but at least one channel OPERABLE. If the channel(s) cannot be restored to operable status within 30 days, a special report is required to be submitted to report the alternate method of monitoring the appropriate parameter(s), cause of the inoperability and plans and schedule for restoring the channel to OPERABLE status. This change eliminates a shutdown requirement for any single redundant channel inoperable.

ACTION Statement b. is modified to address no operable channel(s) for one or more instrument functions. It requires one channel to be restored to operable status within 7 days, versus the CTS 48 hour requirement, otherwise enter the Action Requirement referenced in Table 3.3-10. All exceptions to the ACTION Statement currently included in the TS are deleted. The requirement for a plant shutdown for an inoperable RVLIS and containment area radiation monitor channels inoperable is eliminated and replaced with a requirement to submit a Special Report.

Revised ACTION Statement c. requires the plant to be in Mode 3 within 6 hours and Mode 4 within the next 6 hours for all Functions except the containment Hydrogen Concentration monitors, and will require that the plant be in MODE 3 within 6 hours for both Containment Hydrogen Concentration monitors inoperable. This ACTION Statement will not be applicable to RVLIS and the containment area radiation monitor high range channels.

New ACTION Statement d. will be applicable to RVLIS and the containment area radiation high range monitor channels and requires a special report to be submitted in accordance with Technical Specification 6.9.2.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS30
(continued)

CTS Table 3.3-10 is modified accordingly.

NUREG-1431 changes (1) the allowed outage time (AOT) for TS 3/4.3.3.6 from 7 days to 30 days for one inoperable channel for one or more functions; (2) the AOT for two channels inoperable from 48 hours to 7 days and (3) allows other relaxations. NUREG-1431 also specifies that all RG 1.97 Type A and Category 1 accident monitoring instrumentation must be included in each plant's TS. Specific Justification of the changes is provided below.

Addition and Deletion of Accident Monitoring Instrumentation

The addition of Type 1 and Category A instrumentation provides assurance that instrumentation which provides indication of the variables considered critical for monitoring plant conditions following a DBA are available during an accident. The removal of instrumentation that is not classified as Type A or Category 1 reduces the chance of an unnecessary shutdown and the associated risks.

The change of the required number of containment recirculation sump and containment area radiation monitoring channels is a conservative change that makes revised ACTION Statement a. applicable to these channels.

Action Statement a.

Replacing the shutdown requirement with a requirement to submit a Special Report will eliminate the risk associated with a unit shutdown, yet still provide assurance that the affected channel will be restored in a timely manner.

Action Statement b.

Increasing the AOT from 48 hours to 7 days for the condition when no accident monitoring instrumentation channels for a particular function are operable provides additional time to diagnose and correct a problem, provides alternate means for monitoring the accident parameter, and could avoid the risk associated with unnecessary plant transients and shutdowns.

Revised Action Statements c. and d.

The revision of the current ACTION Statements c. and d. is an administrative change that eliminates confusing Action Statements by deleting the numerous exceptions to the action statements. The shutdown and special report requirements previously contained in these ACTION Statements are revised per NUREG-1431.

The additional Type A and/or Category 1 instrumentation provides assurance that all necessary information is available to operators to allow for proper mitigation of the consequences of an accident.

Although non-Type A/non-Category 1, instrumentation may be useful in determining plant conditions, it is not necessary to include these instruments in the TS. An evaluation of the non-Type A instrumentation in TS 3/4.3.3.6 was performed in accordance with 10CFR50.36(c)(2)(ii). A probabilistic risk assessment (PRA) evaluation for DCPD indicated that instruments which are non-Type A in TS 3/4.3.3.6 did not contain constraints or prime importance in limiting the likelihood or severity of accident sequences which are commonly found to dominate offsite health effects and do not significantly contribute to risk.

The specific criteria used to determine if an accident sequence was risk dominant for core damage frequency was any sequence that had a probability of occurring greater than 1E-6 per reactor year (as a conservative initial determination). For off-site health effects, any sequence whose frequency of serious radioactive releases is commonly found to be greater than 1E-7 per reactor year, was considered to be a dominant risk. The following non-Type A instrumentation is currently included in TS 3/4.3.3.6:

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS30 (continued)

- Reactor Coolant System Subcooling Margin Monitor (B2)
- PORV Position Indicator (D2)
- PORV Block Valve Position Indicator (D2)
- Safety Valve Position Indicator (D2)
- Main Steam Line Radiation Monitor (D2)
- Plant Vent Area Radiation Monitor - High Range (E2)
- Reactor Vessel Level Indication System (B1)
- Containment Area Radiation Monitor - High Range (E1)

Although the non-Type A equipment was determined not to be risk significant, because of the information provided, it was determined that non-Type A/Category 1 equipment should remain in the TS. Non-Type A/non-Category 1 equipment that is being deleted from the TS is proposed to be relocated to plant Equipment Control Guidelines.

Action Statement a.

Increased AOT

Two channels provide indication for the Type A instrumentation included in proposed TS 3/4.3.3.6, except for containment isolation valve (CIV) position, auxiliary feedwater (AFW) flow and steam generator (SG) water level (wide range) channels. Loss of one of two channels would not cause a loss of function. The other channel would still be available to provide indication of the parameter. Increasing the AOT from 7 days to 30 days could avoid a shutdown and the associated risks while still providing assurance that indication of the parameter is available while assuring that an out of service channel is returned to service in a timely manner.

For the CIV position, the important information is the status of the containment penetrations. The TS requires one position indicator for each active containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of passive valve or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status.

Neither the AFW flow rate indication nor the SG water level wide range channels are redundant. The purpose of the AFW flow indication channel is to provide indication of AFW flow provided to the SG so that adequate decay heat removal capability from the RCS via the SGs can be verified. Similarly, the SG water level channel indicates the water level in the SGs so that adequate decay heat removal from the RCS via the SGs can be verified. Both the AFW flow indication and the SG water level wide range channels provide verification of decay heat removal. Since both the AFW flow and SG level channels monitor decay heat removal capability in a different way, they serve as diverse indications of the same parameter. Consequently, the loss of either channel would not prevent an operator from monitoring the decay heat removal capability from the RCS via each SG.

Since either redundant or diverse means of indication exist for each Type A channel, the monitoring of the function will not be lost as a result of one channel failure, and increasing the AOT from 7 days to 30 days will not have a significant affect on the health and safety of the public. Since the non-Type A/Category 1 instrumentation does not significantly affect risk, increasing the AOT from 7 days to 30 days for this instrumentation will not have a significant affect on the risk to the health and safety of the public.

Elimination of Shutdown Requirement from ACTION a.

The removal of the shutdown requirement could avoid a plant shutdown and the associated risks. The reporting requirements specify that alternate means of monitoring these parameters and a

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS30 (continued)

schedule for repairing the channels be submitted to the NRC. This reporting requirement will provide assurance that the channels will be repaired in a timely manner. As a result of the alternate means of monitoring the affected parameter, and the redundant channel available, the required function will not be lost during the time when one channel is inoperable.

Action Statement b.

The 7 day AOT for the complete loss of monitoring for a function is allowed because of the relatively low probability of an event requiring the accident monitoring instrumentation and the availability of alternate means to obtain the required information (diverse instrumentation or other non-Regulatory Guide 1.97 instrument channels to monitor the function).

Additionally, the probability of an accident occurring during the time when both channels are out of service is small. The probability of a LOCA requiring the operability of post accident monitoring channels is $1E-6$ per reactor year.

For all channels except the high range containment area radiation monitor and RVLIS, a shutdown is required if the AOT is exceeded. This assures that, for the complete loss of a monitoring function that is deemed risk significant, the plant is placed in a condition in which the function is no longer required in a timely manner.

The shutdown requirement for both the high range containment area radiation monitor and the RVLIS are proposed to be replaced with a reporting requirement. These channels are non-Type A channels. Since these channels were identified as non-risk significant, removing the shutdown requirement for these channels will have no significant negative impact on safety. The reporting requirements specify that alternate means of monitoring these parameters and a schedule for repairing the channels be submitted to the NRC. This reporting requirement will provide assurance that the channels will be repaired in a timely manner.

Current Action Statements c. and d.

The revision of the current Action Statements c. and d. is an administrative change.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS30 (continued)

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

None of the changes alter the plant configuration or operation. Consequently, this change does not significantly increase the probability of an accident as defined in the FSAR Update. Additionally, no post accident monitoring channel initiates an accident. Except for the increased instrument allowed outage time, the change is essentially a reformatting and deletion of obsolete statements. The evaluation identifies that there is a low probability of an event which would require the accident monitoring instrumentation and that there are alternate means to obtain the required information if the accident monitoring function is required. Consequently, the change does not have a significant affect on the probability or consequences of any previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

None of the changes require physical alteration to any plant system or change the method by which any safety-related system performs its function. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

None of the changes will change any accident analysis assumptions, initial conditions, or results. Consequently, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS30" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c) and; accordingly, an NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS35
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The Applicable Modes or Other Specified Conditions in the CTS include Mode 4 for the manual initiation of Steam Line Isolation. However, neither the CTS 3.7.1.5 Main Steam Isolation Valves nor NUREG-1431 ITS 3.7.1.2 requires the valves to be OPERABLE in MODE 4. In MODE 4, the Steam Generator energy is low and requiring the manual initiation circuitry to be OPERABLE without the valves being OPERABLE serves no purpose. Therefore, the MODE 4 OPERABILITY requirement has been deleted.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The CTS do not require the MSIVs to be OPERABLE in MODE 4. There is no change in the Operability requirements of the MSIVs; therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in Applicability will not affect the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS35" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c) and; accordingly, an NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS39
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

APPLICABILITY Note [*] and ACTION Statements [11] for Functional Units [1, 6.b, and 20] of current TS Table 3.3-1 are modified to provide an alternative to opening the reactor trip breakers (RTBs) while still assuring that the function and intent of opening the RTBs is met. As currently worded, these ACTION Statements result in a feedwater isolation signal (FWIS) when in MODE 3 with a T_{avg} less than [554°F. FSAR Table 7.3-13 and FSAR Figure 7.2-1 (sht. 13) detail the FWIS generation on the coincidence of P-4 and low T_{avg} .] A more generic action, which assures the rods are fully inserted and cannot be withdrawn, replaces the specific method of precluding rod withdrawal. The revised Applicability and ACTION Statements still assure rod withdrawal is precluded. This change does not involve any safety impact and is consistent with Traveler TSTF-135. The proposed change allows more freedom in how rod withdrawal is precluded and is thus less restrictive. However, the intent of using physical plant characteristics to prevent rod withdrawal is not diminished. The specification now acknowledges that the Rod Control System can be effectively disabled by means other than opening the RTBs.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not alter the requirement to preclude rod withdrawal using physical plant characteristics. The specification does not allow administrative control or other means which could be conceived as less stringent. The specification does allow for alternative means to opening the RTBs for precluding rod withdrawal. These means, if used, would be as effective as opening the RTBs, such as removing power to the Rod Control System. Therefore, there should be no increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Inadvertent rod withdrawal accidents have been previously evaluated. This change does not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS39 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will preclude rod withdrawal with the same level of assurance that opening of the RTBs provided. No reduction in the margin of safety will result from this change.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS39" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c) and; accordingly, an NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS40
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS require that the setpoints be verified during quarterly TADOTs for the Reactor Coolant Pump (RCP) Underfrequency [and RCP Undervoltage] functions. The setpoint is adequately confirmed during the 18 month calibrations.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The setpoint for the RCP Underfrequency [and RCP Undervoltage functions are] selected such that the Allowable value is not expected to be exceeded during the 18 months between calibrations. This function is not an initiator for an accident previously analyzed but the function is used to provide a reactor trip signal to mitigate certain accidents. However, because the function is not expected to exceed its Allowable Value bases on the 18 month calibration, it is concluded that the proposed change would not involve a significant increase in the consequences of any accident previously analyzed.

2. Does the change create the possibility of a new or different kind of accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change will not alter the normal method of plant operation. No new transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS40 (continued)

nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS40" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a NSHC finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG.-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS CONTENTS

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Mark-up

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NUREG-1431 SPECIFICATIONS WHICH ARE NOT APPLICABLE

3.3.9 Boron Dilution Protection System

INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.3

<u>TRAVELER #</u>	<u>STATUS</u>	<u>DIFFERENCE #</u>	<u>COMMENTS</u>
TSTF-19, Rev. 1	Not Incorporated	NA	Not NRC approved as of traveler cut-off date.
TSTF-36, Rev. 2	Incorporated	3.3-34	
TSTF-37, Rev. 1	Not Incorporated	NA	ITS 5.6.8 still addresses PAM reports. Sections after ITS 5.6.7 were not renumbered.
TSTF-51	Not Incorporated	NA	Requires plant-specific reanalysis to establish decay time dependence for fuel handling accident.
TSTF-91	Not Incorporated	NA	[Trip Setpoints and] Allowable Values for loss of voltage and degraded voltage will remain in the TS.
TSTF-111, Rev. 1	Incorporated	NA	
TSTF-135	Partially Incorporated	3.3-41, 3.3-93, 3.3-95, 3.3-122	Traveler is too broad scope in nature; should have been separate travelers. Portions of the traveler that significantly clarify operability requirements have been incorporated.
TSTF-161	Incorporated	3.3-79	
TSTF-168	Incorporated	3.3-43	
TSTF-169	Incorporated	3.3-42	
WOG-106	Incorporated	3.3-49	
Proposed Traveler	Incorporated	3.3-107	WOG Mini-group Action Item # 45

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u> B.2-1 Be in MODE 3.	54 hours
	<u>AND</u> B.2.2 Open reactor trip breakers (RTBs).	<u>3.3-106</u> 55 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>NOTE While this LCO is not met for function 19, 20 or 21 entry into MODE 5^(b) from MODE 5 is not permitted This NOTE is an exception to the requirements of LCO 3.0.4</p> <p>C. One channel or train inoperable.</p>	<p>C.1 Restore channel or train to OPERABLE status.</p> <p><u>OR</u></p> <p>C.2¹ Open RTBs. Fully insert all rods.</p> <p><u>AND</u></p> <p>C.2² Place the Rod Control System in a condition incapable of rod withdrawal.</p>	<p style="text-align: right;"><u>3.3-135</u></p> <p>48 hours</p> <p>49 hours <u>3.3-122</u></p> <p>49 hours <u>3.3-122</u></p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. One Power Range Neutron Flux-High channel inoperable.</p>	<p>-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. -----</p>	
	<p>D.1.1 Place channel in trip.</p>	<p>6 hours</p>
	<p><u>AND</u></p>	<p>12 hours</p>
	<p>D.1.2 Reduce THERMAL POWER to ≤ 75% RTP.</p>	<p>6 hours</p>
	<p><u>OR</u> D.2.1 Place channel in trip. <u>AND</u></p>	<p>(continued)</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	<p>-----NOTE----- Only NOT required to be performed when the unit is 12 hours after input from one Power Range Neutron Flux input channel to QPTR is inoperable and THERMAL POWER is > than 75% RTP.</p> <p>D.2.2 Perform SR 3.2.4.2.</p> <p><u>OR</u></p> <p>D.3 Be in MODE 3.</p>	<p><u>3.3-120</u></p> <p>Once per 12 hours</p> <p>12 hours</p>
E. One channel inoperable.	<p>-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels.</p> <p>E.1 Place channel in trip.</p> <p><u>OR</u></p> <p>E.2 Be in MODE 3.</p>	<p><u>3.3-40</u></p> <p>6 hours</p> <p>12 hours</p>
F. THERMAL POWER > P-6 and < P-10. One Intermediate Range Neutron Flux channel inoperable.	<p>F.1 Reduce THERMAL POWER to < P-6.</p> <p><u>OR</u></p> <p>F.2 Increase THERMAL POWER to > P-10.</p>	<p>224 h <u>3.3-95</u></p> <p>ours <u>3.3-107</u></p> <p>224 h <u>3.3-107</u></p> <p>ours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G. THERMAL POWER > P-6 and < P-10, Two Intermediate Range Neutron Flux channels inoperable.</p>	<p>G.1 Suspend operations involving positive reactivity additions.</p> <p><u>AND</u></p> <p>G.2 Reduce THERMAL POWER to < P-6.</p>	<p>Immediately</p> <p style="text-align: center;"><u>3.3-95</u></p> <p>2 hours</p>
<p>H. NOT USED THERMAL POWER < P-6, one or two Intermediate Range Neutron Flux channels inoperable.</p>	<p>H.1 Restore channel(s) to OPERABLE status.</p>	<p>Prior to <u>3.3-95</u> increasing THERMAL POWER to > P-6</p>
<p>I. One Source Range Neutron Flux channel inoperable.</p>	<p>I.1 Suspend operations involving positive reactivity additions.</p>	<p>Immediately</p>
<p>J. Two Source Range Neutron Flux channels inoperable.</p>	<p>J.1 Open RTBs.</p>	<p>Immediately</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. One Source Range Neutron Flux channel inoperable.	K.1 Restore channel to OPERABLE status.	48 hours
	OR	
	K.2.1 Open RTBs. Fully insert all rods. AND K.2.2 Place the Control Rod System in a condition incapable of rod withdrawal.	49 hours <u>3.3-122</u> 49 hours <u>3.3-122</u>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>L. Required Source Range Neutron Flux channel (S) inoperable.</p>	<p>L.1 Suspend operations involving positive reactivity additions.</p> <p><u>AND</u></p> <p>L.2 Close unborated water source isolation valves.</p> <p><u>AND</u></p> <p>L.32 Perform SR 3.1.1.1.</p>	<p>Immediately</p> <p style="text-align: right;"><u>B</u></p> <p>1 hour <u>3.3-123</u></p> <p>1 hour</p> <p><u>AND</u></p> <p>Once per 12 hours thereafter</p>
<p>M. One channel inoperable.</p>	<p>-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. -----</p> <p>M.1 Place channel in trip.</p> <p><u>OR</u></p> <p>M.2 Reduce THERMAL POWER to < P-7.</p>	<p>6 hours</p> <p>12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>N. NOT USED One Reactor Coolant Flow Low (Single Loop) channel inoperable.</p>	<p>NOTE</p> <p>The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.</p> <hr/> <p>N.1 Place channel in trip.</p> <p>OR</p> <p>N.2 Reduce THERMAL POWER to < P-8.</p>	<p>3.3-42</p> <p>6 hours</p> <p>10 hours</p>
<p>O. NOT USED One Reactor Coolant Pump Breaker Position channel inoperable.</p>	<p>NOTE</p> <p>The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.</p> <hr/> <p>O.1 Restore channel to OPERABLE status.</p> <p>OR</p> <p>O.2 Reduce THERMAL POWER to < P-8.</p>	<p>3.3-103</p> <p>6 hours</p> <p>10 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>P. One or more Turbine Trip channel(s) inoperable.</p>	<p>-----NOTE----- The inoperable low auto stop of pressure channel may be bypassed for up to 4 hours for surveillance testing of other channels. -----</p> <p>P.1 Place channel(s) in trip.</p> <p><u>OR</u></p> <p>P.2 Reduce THERMAL POWER to < P_{sg}.</p>	<p><u>3.3-02</u></p> <p>6 hours</p> <p><u>B</u></p> <p>10 hours</p>
<p>Q. One train inoperable.</p>	<p>-----NOTE----- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. -----</p> <p>Q.1 Restore train to OPERABLE status.</p> <p><u>OR</u></p> <p>Q.2 Be in MODE 3.</p>	<p><u>B</u></p> <p>6 hours</p> <p>12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>R. One RTB train inoperable.</p>	<p>-----NOTES-----</p> <p>1. One train may be bypassed for up to 2 hours for surveillance testing or maintenance, provided the other train is OPERABLE.</p> <p>2. One RTB may be bypassed only for up to 2 hours the time required for performing maintenance on undervoltage or shunt trip mechanisms per CONDITION U, provided the other train is OPERABLE.</p> <p>3. One RTB may be bypassed for up to 4 hours for logic testing per CONDITION Q, provided the other train is OPERABLE.</p> <p>-----</p> <p>R.1 Restore train to OPERABLE status.</p> <p><u>OR</u></p> <p>R.2 Be in MODE 3.</p>	<p><u>3.3-43</u></p> <p><u>3.3-43</u> ⁿ</p> <p><u>3.3-117</u></p> <p><u>3.3-117</u></p> <p><u>3.3-03</u></p> <p>1 hour</p> <p>7 hours</p>
<p>S. One or more required channels or trains inoperable.</p>	<p>S.1 Verify interlock is in required state for existing unit conditions.</p> <p><u>OR</u></p> <p>S.2 Be in MODE 3.</p>	<p>1 hour <u>3.3-44</u></p> <p>7 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>T. One or more required channels or trains inoperable.</p>	<p>T.1 Verify interlock is in required state for existing unit conditions.</p> <p><u>OR</u></p> <p>T.2 Be in MODE 2.</p>	<p>1 hour <u>3.3-44</u></p> <p>7 hours</p>
<p>U. One trip mechanism inoperable for one RTB.</p>	<p>U.1 Restore inoperable trip mechanism to OPERABLE status.</p> <p><u>OR</u></p> <p>U.2-1 Be in MODE 3.</p> <p><u>AND</u></p> <p>U.2-2 Open RTB.</p>	<p>48 hours</p> <p>54 hours <u>3.3-106</u></p> <p>55 hours</p>
<p>V. NOT USED Two RTS trains inoperable.</p>	<p>V.1 Enter LCO 3.0.3.</p>	<p>Immediately <u>3.3-93</u></p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>W One channel inoperable</p>	<p>NOTE The inoperable channel may be bypassed for up to 72 hours for surveillance or maintenance.</p> <p>W.1 Place channel in trip</p>	<p>6 hours <u>3.3-45</u></p>
<p>X One or more SG-low low Trip Time Delay circuit delay timers inoperable</p>	<p>X.1 Adjust the Trip Time Delay threshold power level for 0 seconds time delay to 0% RTP</p> <p>OR</p> <p>X.2 Place the affected SG-low low level in trip.</p>	<p>6 hours <u>3.3-46</u></p> <p>6 hours</p>

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	<p>-----NOTES-----</p> <ol style="list-style-type: none"> Adjust NIS channel if absolute difference is > 2%. Not required to be performed until [12] 24 hours after THERMAL POWER is \geq 15% RTP but prior to exceeding 30% RTP. <p>-----</p> <p>Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.</p>	<p style="text-align: right;"><u>B-PS</u> <u>3.3-47</u></p> <p>24 hours</p>
SR 3.3.1.3	<p>-----NOTES-----</p> <ol style="list-style-type: none"> Adjust NIS channel if absolute difference is \geq 3%. Not required to be performed until 24 hours after THERMAL POWER is \geq [15] 50% RTP. <p>-----</p> <p>Compare results of the incore detector measurements to NIS AFD.</p>	<p style="text-align: right;"><u>B</u> <u>3.3-96</u></p> <p>31 effective full power days (EFPD)</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.4 -----NOTE----- This Surveillance must be performed on the reactor trip bypass breaker for the local manual shunt trip only prior to placing the bypass breaker in service. ----- Perform TADOT:</p>	<p style="text-align: center;"><u>3.3-124</u></p> <p>31 days on a STAGGERED TEST BASIS</p>
<p>SR 3.3.1.5 Perform ACTUATION LOGIC TEST.</p>	<p>31 days on a STAGGERED TEST BASIS</p>
<p>SR 3.3.1.6 -----NOTE----- Not required to be performed until 24 hours after achieving equilibrium conditions with THERMAL POWER ≥ 50 75% RTP. ----- Calibrate excore channels to agree with incore detector measurements.</p>	<p style="text-align: center;"><u>B</u></p> <p style="text-align: center;"><u>3.3-06</u></p> <p>92 EFPD <u>B</u></p>
<p>SR 3.3.1.7 -----NOTE----- 1 Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. 2 For source range instrumentation, this Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. ----- Perform COT.</p>	<p style="text-align: center;"><u>B</u></p> <p style="text-align: center;"><u>3.3-111</u></p> <p>92 day <u>B</u> S</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.8</p> <p>-----NOTE----- This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. -----</p> <p>Perform COT.</p>	<p>-----NOTE----- Only required when not performed within <u>B</u> previous 92 days -----</p> <p>Prior to reactor startup</p> <p><u>AND</u> <u>3.3-49</u></p> <p>Four¹² hours after reducing power below P-10 for power and intermediate instrumentation</p> <p><u>AND</u></p> <p>Four hours after reducing power below P-6 for source range instrumentation</p> <p><u>AND</u></p> <p>Every 92 days thereafter</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	<p>-----NOTE----- Verification of setpoint is not required. -----</p> <p>Perform TADOT.</p>	<p>92 day <u> </u> S <u> </u> <u>B</u></p>
SR 3.3.1.10	<p>-----NOTE----- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -----</p> <p>Perform CHANNEL CALIBRATION.</p>	<p>18 months <u> </u> <u> </u> <u>B</u></p>
SR 3.3.1.11	<p>-----NOTES-----</p> <p>1. Neutron detectors are excluded from CHANNEL CALIBRATION.</p> <p>2. This Surveillance shall include verification that the time constants are adjusted to the prescribed values.</p> <p>3. Power and Intermediate Range detector plateau voltage verification is not required to be performed prior to entry in to MODE 2 or 1.</p> <p>-----</p> <p>Perform CHANNEL CALIBRATION.</p>	<p><u> </u> 3.3-125</p> <p><u> </u> 3.3-07</p> <p>18 months <u> </u> <u> </u> <u>B</u></p>
SR 3.3.1.12	<p>-----NOTE-----</p> <p>This Surveillance shall include verification of Reactor Coolant System resistance temperature detector bypass loop flow rate.</p> <p>-----</p> <p>Perform CHANNEL CALIBRATION.</p>	<p><u> </u> 3.3-101</p> <p>18 months <u> </u> <u> </u> <u>B</u></p>
SR 3.3.1.13	Perform COT.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
(continued)	
SR 3.3.1.14 -----NOTE----- Verification of setpoint is not required. ----- Perform TADOT.	18 months <u> B </u>
SR 3.3.1.15 -----NOTE----- Verification of setpoint is not required. ----- Perform TADOT.	-----NOTE----- Only required when not performed within previous 31 days ----- Prior to reactor startup
SR 3.3.1.16 -----NOTE----- Neutron detectors are excluded from response time testing. ----- Verify RTS RESPONSE TIME is within limits.	18 months <u> B </u> on a STAGGERED TEST BASIS
SR 3.3.1.17 Perform ACTUATION LOGIC TEST	18 months <u> 3.3-45 </u>

Table 3.3.1-1 (page 1 of 810)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT ^(a)
1. Manual Reactor Trip	1.2	2	B	SR 3.3.1.14	NA	NA
	3 ^(b) , 4 ^(b) , 5 ^(b)	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						<u>B-PS</u>
a. High	1.2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ [111-2] 111% RTP	≤ 100% RTP <u>B</u>
b. Low	1 ^(c) , 2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ [27-2] 27% RTP	≤ 25% RTP <u>B</u>
3. Power Range Neutron Flux Rate						<u>B-PS</u>
a. High Positive Rate	1.2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ [6-8] 6% RTP with time constant ≥ 2 sec	≤ 5% RTP with time constant ≥ 2 sec
b. High Negative Rate	1.2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ [6-8] 6% RTP with time constant ≥ 2 sec	≤ 5% RTP with time constant ≥ 2 sec <u>B</u>
4. Intermediate Range Neutron Flux	1 ^(c) , 2 ^(d)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ [31] 30-9% RTP	≤ 25% RTP <u>B-PS</u>
	2 ^(e)	2	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ [31] % RTP	≤ 25% RTP <u>3.3-95</u>

(continued)

- (a) ~~Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~ ED
- (b) ~~With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal on all rods not fully inserted.~~ 3.3-122

- (c) Below the P-10 (Power Range Neutron Flux) interlocks.
- (d) Above the P-6 (Intermediate Range Neutron Flux) interlocks.
- ~~(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.~~

3.3-95

Table 3.3.1-1 (page 2 of 810)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SET POINT	ED
						NT (a)	
							<u>B</u>
5. Source Range Neutron Flux	2(e)	2	I, J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 1.4E5 cps		
	3 ^(b) , 4 ^(b) , 5 ^(b)	2	J, K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 1.4E5 cps	≤ 1.0E5 cps	
	3 ^(f) , 4 ^(f) , 5 ^(f)	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	≤ 1.0E5 cps	
							N/A
6. Overtemperature ΔT	1.2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16	Refer to Note 1 (Page 3.3-214)	Refer to Note 1 (Page 3.3-214)	
							<u>3.3-101</u>
7. Overpower ΔT	1.2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.12 SR 3.3.1.16	Refer to Note 2 (Page 3.3-225)	Refer to Note 2 (Page 3.3-225)	
							<u>3.3-101</u>

(continued)

(a) ~~Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

ED

(b) ~~With RTBs closed and Rod Control System capable of rod withdrawal or all rods not fully inserted.~~

3.3-122

(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(f) ~~With the RTBs open or all rods fully inserted and incapable of withdrawal. In this condition, source range Function does not provide reactor trip but does provide [input to the Boron Dilution Protection System (LCO 3.3.9), and] indication.~~

3.3-11

B-PS

Table 3.3.1-1 (page 3 of 810)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT (a)
8. Pressurizer Pressure						B-PS
a. Low	1(g)	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	$\geq [1886]$ 1944 psig	$\geq [1900]$ 1950 psig B
b. High	1.2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	$\leq [2396]$ 2390 psig	
9. Pressurizer Water Level - High	1(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	$\leq [93.8]$ 92.5%	\leq B-
10. Reactor Coolant Flow - Low	1(g)	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	$\geq [89.7]$ %	$\geq [90]$ % B- 3.3-09 3.3-42
a. Single Loop	1(h)	3 per loop	H	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	$\geq [89.2]$ %	3.3-42
b. Two Loops	1(i)	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	$\geq [89.2]$ %	$\geq [90]$ %

(continued)

(a) ~~Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

ED

(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

(h) ~~Above the P-8 (Power Range Neutron Flux) interlock.~~

3.3-42

(i) ~~Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.~~

3.3-42

(j) ~~Minimum measured flow is 89,800 gpm per loop for Unit 1 and 90,625 gpm per loop for Unit 2.~~

3.3-09

Table 3.3.1-1 (page 4 of 810)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT (a)
<u>3.3-103</u>						
11. Reactor Coolant Pump (RCP) Breaker Position	1 (g)	1 per RCP	M	SR 3.3.1.14	NA	NA
a. Single Loop	1 (h)	1 per RCP	0	SR 3.3.1.14	NA	NA
b. Two Loops	1 (i)	1 per RCP	M	SR 3.3.1.14	NA	NA
<u>B-PS</u>						
12. Undervoltage RCPs	1 (g)	2 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ [4760] 7730 V each bus	≥ [4830] 8050 V each bus
<u>B-PS</u>						
<u>B</u>						
13. Underfrequency RCPs	1 (g)	3 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ [57.1] 53.9 Hz each bus	≥ [57.5] 54.0 Hz each bus
<u>B-PS</u>						
14. Steam Generator (SG) Water Level - Low Low	1.2	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ [30.4] 6.8%	≥ [32.3] 7.2%
<u>3.3-46</u>						
Coincident with						
a) RCS Loop at equivalent to power ≤ 50% RTP with a time delay (TD)	1.2	4 (1/loop)	X	SR 3.3.1.7 SR 3.3.1.10	RCS loop at variable input ≤ 51.5% RTP	RCS loop at variable input ≤50%
OR						
b) RCS Loop at equivalent to power > 50% RTP with no time delay.	1	4(1/loop)	X	SR 3.3.1.7 SR 3.3.1.10	RCS loop at variable input > 51.5 RTP TD=0	RCS loop at variable input > 50% RTP TD=0

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT (a)
						<u>3.3-01</u>
15. NOT USED SG Water Level Low	1.2	2 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ [30.4]%	≥ [32.3]%
Coincident with Steam Flow Feedwater Flow Mismatch	1.2	2 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ [42.5]% full steam flow at RTP	≤ [40]% full steam flow at RTP

(continued)

~~(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

~~(g) Above the P-7 (Low Power Reactor Trips Block) interlock.~~

~~(h) Above the P-8 (Power Range Neutron Flux) interlock.~~

3.3-103

~~(i) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.~~

3.3-103

Table 3.3.1-1 (page 55 of 810)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT (a)
<u>B-PS</u>						
16. Turbine Trip						
a. Low Fluid Auto-Stop Oil Pressure	1(j)	3	P	SR 3.3.1.10 SR 3.3.1.15	≥ [750] 45 psig	≥ [800] 50 psig <u>B</u> open
b. Turbine Stop Valve Closure	1(j)	4	P	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open	
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1.2	2 trains	Q	SR 3.3.1.14	NA	NA

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT (a)
18. Reactor Trip System Interlocks						B
a. Intermediate Range Neutron Flux, P-6	2(e)	2	S	SR 3.3.1.11 SR 3.3.1.13	≥ 6E-11 amp	≥ 1E-10 amp 3.3-54
b. Low Power Reactor Trips Block, P-7	1	1 per train	T	NA SR 3.3.1.11 SR 3.3.1.13	NA	NA 3.3-44 B-PS
c. Power Range Neutron Flux, P-8	1	43	T	SR 3.3.1.11 SR 3.3.1.13	≤ [50-2] 37.1% RTP	≤ [48] 35% RTP 3.3-44 B-PS B
d. Power Range Neutron Flux, P-9	1	43	T	SR 3.3.1.11 SR 3.3.1.13	≤ [52-2] 52.1% RTP	≤ 50% RTP B-PS B 3.3-44
e. Power Range Neutron Flux, P-10	1,2	43	S	SR 3.3.1.11 SR 3.3.1.13	≥ [7-8] 7.9% RTP and ≤ [12-2] 12.1% RTP	≥ 10% RTP B B-PS PS
f. Turbine Impulse Chamber Pressure, P-13	1	2	T	SR 3.3.1.11 SR 3.3.1.10 SR 3.3.1.13	≤ [12-2] 12.1% RTP turbine impulse pressure equivalent	≤ 10% RTP turbine impulse pressure equivalent

(a) ~~Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(j) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 68 of 810)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	<u>E</u>
						TRIP SETPOINT ^(a)
19. Reactor Trip Breakers ^(k)	1.2	2 trains	R	SR 3.3.1.4	NA	NA
	3 ^(b) , 4 ^(b) , 5 ^(b)	2 trains	C	SR 3.3.1.4	NA	NA
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms ^(k)	1.2	1 each per RTB	U	SR 3.3.1.4	NA	<u>3.3-124</u>
	3 ^(b) , 4 ^(b) , 5 ^(b)	1 each per RTB	C	SR 3.3.1.4	NA	NA
21. Automatic Trip Logic	1.2	2 trains	Q	SR 3.3.1.5	NA	NA
	3 ^(b) , 4 ^(b) , 5 ^(b)	2 trains	C	SR 3.3.1.5	NA	NA
						<u>3.3-45</u>
22. Seismic Trip	1.2	3 direction (X, Y, Z) in 3 locations	W	SR 3.3.1.12 SR 3.3.1.14 SR 3.3.1.17	≤ 0.40 _s	≤ 0.35 _s

(a) ~~Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(b) ~~With RTBs closed and Rod Control System capable of rod withdrawal on all rods not fully inserted.~~

(k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

ED

3.3-122

Table 3.3.1-1 (page 79 of 810)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than ~~3.8~~ 1.0% of ΔT span.

B-PS

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} \leq \Delta T_O \left\{ K_1 - K_2 \left[\frac{(1 + \tau_1 s)}{(1 + \tau_2 s)} T - T' \right] + K_3 (P - P') - f_1(\Delta T) \right\}$$

3.3-10

Where:

- ΔT is measured RCS ΔT , °F.
- ΔT_0 is the indicated ΔT at RTP, °F.
- s is the Laplace transform operator, sec⁻¹.
- T is the measured RCS average temperature, °F.
- T' is the nominal T_{avg} at RTP, ~~588~~ 576.6 (Unit 1) & 577.6 (Unit 2) °F;

3.3-13

- P is the measured pressurizer pressure, psig
- P' is the nominal RCS operating pressure, ~~2235~~ 2235 psig

B-PS

ED

B

- $K_1 \leq \text{[1.09]}$ ~~20~~ $K_2 \geq \text{[0.0138]}$ ~~0.0182~~ /°F $K_3 = \text{[0.000671]}$ ~~0.000831~~ /psig
- $\tau_1 \geq \text{[8]}$ 30 sec $\tau_2 \geq \text{[3]}$ 4 sec $\tau_3 \leq \text{[2]}$ sec
- $\tau_4 \geq \text{[33]}$ 0 sec $\tau_5 \leq \text{[4]}$ 0 sec $\tau_6 \leq \text{[2]}$ sec

B-PS

3.3-10

B

- $f_1(\Delta T) =$ ~~0.0126~~ ~~0.0275~~ (35 19 + (q_t - q_b)) when q_t - q_b ≤ ~~35~~ 19% RTP
- 0% of RTP when ~~35~~ 19% RTP < q_t - q_b ≤ 7% RTP
- ~~0.0105~~ ~~0.0238~~ ((q_t - q_b) - 7) when q_t - q_b > 7% RTP

B-PS

3.3-10

B

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and q_t + q_b is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 810 of 810)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than ~~{3}~~ 10% of ΔT span.

~~_____~~ B-PS
3.3-10

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_3 s)} \leq \Delta T_0 \left\{ K_4 - K_5 \frac{\tau_3 s}{1 + \tau_3 s} T - K_6 [T - T'] - f_2(\Delta T) \right\}$$

~~_____~~ 3.3-13

Where: ΔT is measured RCS ΔT , °F.
 ΔT_0 is the indicated ΔT at RTP, °F.
 s is the Laplace transform operator, sec⁻¹.
 T is the measured RCS average temperature, °F.
 T' is the nominal T_{avg} at RTP, ~~{588}~~ 576.6 (Unit 1) & 577.6 (Unit 2) °F.

$K_4 \leq \{1.09\}$ ~~1.072~~ $K_5 \geq \{0.02\}$ ~~0.0174~~ /°F for increasing T_{avg} $K_6 \geq \{0.00128\}$ ~~0.00145~~ /°F when $T > T'$

$\tau_1 \geq \{8\}$ sec $\tau_2 \leq \{3\}$ sec $\tau_3 \leq \{2\}$ ~~10~~ sec
 $\tau_{d1} \leq \{2\}$ ~~10~~ sec $\tau_{d2} \geq \{10\}$ ~~10~~ sec $\tau_4 \leq \{2\}$ ~~10~~ sec

$f_2(\Delta T) = 0\%$ RTP for all ΔT .

Note 3: Steam Generator Water Level Low-Low Trip Time Delay

~~_____~~ B-PS
3.3-13
3.3-10

$$TD = B1(P)^3 + B2(P)^2 + B3(P) + B4$$

Where: $P =$ RCS Loop ΔT Equivalent to Power (%RTP), $P \leq 50\%$ RTP

$TD =$ Time delay for Steam Generator Water Level Low-Low Reactor Trip (in seconds)

- ~~_____~~ $B1 = 0.007128$
- ~~_____~~ $B2 = 0.8099$
- ~~_____~~ $B3 = 31.40$
- ~~_____~~ $B4 = +464.1$

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or train(s).	Immediately
B. One channel or train inoperable.	B.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u> B.2.1 Be in MODE 3.	54 hours
	<u>AND</u> B.2.2 Be in MODE 5.	84 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. One train inoperable.</p>	<p>C.1 -----NOTE----- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----- Restore train to OPERABLE status. <u>OR</u> C.2.1 Be in MODE 3. <u>AND</u> C.2.2 Be in MODE 5.</p>	<p style="text-align: right;"><u><u>B</u></u></p> <p>6 hours</p> <p>12 hours</p> <p>42 hours</p>
<p>D. One channel inoperable.</p>	<p>D.1 -----NOTE----- The inoperable channel or one additional channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----- Place channel in trip. <u>OR</u> D.2.1 Be in MODE 3. <u>AND</u> D.2.2 Be in MODE 4. <u>AND</u> D.2.3 Be in MODE 5 for Function 1. c.</p>	<p style="text-align: right;"><u><u>3.3-37</u></u></p> <p style="text-align: right;"><u><u>B</u></u></p> <p>6 hours</p> <p>12 hours</p> <p>18 hours</p> <p style="text-align: right;"><u><u>3.3-66</u></u></p> <p>42 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. One Containment Pressure channel inoperable.</p>	<p>E.1 -----NOTE----- One additional channel may be bypassed for up to 4 hours for surveillance testing. ----- Place channel in bypass. <u>OR</u> E.2.1 Be in MODE 3. <u>AND</u> E.2.2 Be in MODE 4. <u>AND</u> E.2.3 Be in MODE 5 for Functions 2.c and 3.b.(3).</p>	<p style="text-align: right;"><u>B</u></p> <p>6 hours</p> <p>12 hours</p> <p>18 hours</p> <p style="text-align: right;"><u>3.3-66</u></p> <p>42 hours</p>
<p>F. One channel or train inoperable.</p>	<p>F.1 Restore channel or train to OPERABLE status. <u>OR</u> F.2.1 Be in MODE 3. <u>AND</u> F.2.2 Be in MODE 4.</p>	<p>48 hours</p> <p>54 hours</p> <p>60 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G. One train inoperable.</p>	<p>G.1 -----NOTE----- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----- Restore train to OPERABLE status.</p>	<p style="text-align: right;"><u>B</u></p> <p>6 hours</p>
	<p><u>OR</u></p>	<p>12 hours</p>
	<p>G.2.1 Be in MODE 3.</p>	<p>18 hours</p>
	<p><u>AND</u> G.2.2 Be in MODE 4.</p>	<p>18 hours</p>
<p>H. One train inoperable.</p>	<p>H.1 -----NOTE----- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----- Restore train to OPERABLE status.</p>	<p style="text-align: right;"><u>B</u></p> <p>6 hours</p>
	<p><u>OR</u> H.2 Be in MODE 3.</p>	<p>12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>I. One channel inoperable.</p>	<p>I.1</p> <p>-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. -----</p> <p>Place channel in trip.</p> <p>OR</p> <p>I.2.1 Be in MODE 3.</p> <p>AND</p> <p>I.2.2 Be in MODE 3 for function 5.b.</p>	<p style="text-align: right;"><u>B</u></p> <p>6 hours</p> <p>12 hours <u>3.3-127</u></p> <p>12 hours</p>
<p>J. NOT USED One Main Feedwater Pumps trip channel inoperable.</p>	<p>J.1 Restore channel to OPERABLE status.</p> <p>OR</p> <p>J.2 Be in MODE 3.</p>	<p>48 hours <u>3.3-116</u></p> <p>64 hours</p>
<p>K. NOT USED One channel inoperable.</p>	<p>K.1 NOTE One additional channel may be bypassed for up to 4 hours for surveillance testing.</p> <p>Place channel in bypass.</p> <p>OR</p>	<p>6 hours</p> <p>(continued)</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>K. (continued)</p>	<p>K.2.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>K.2.2 Be in MODE 5.</p>	<p>12 hours</p> <p>42 hours</p>
<p>L. One or more required channels or train inoperable.</p>	<p>L.1 Verify interlock is in required state for existing unit condition.</p> <p><u>OR</u></p> <p>L.2.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>L.2.2 Be in MODE 4.</p>	<p>1 hour <u>3.3-44</u></p> <p>7 hours</p> <p>13 hours</p>
<p>M. One RCS Loop Delta channel inoperable.</p>	<p>M.1 Adjust the Trip Time Delay threshold power level for zero seconds time delay to 0% RTP.</p> <p><u>OR</u></p> <p>M.2 Place the affected SG water level low-low channel in trip.</p>	<p>6 hours <u>3.3-46</u></p> <p>6 hours</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>N: One channel inoperable</p>	<p>N:1 Restore channel to OPERABLE status</p> <p>OR</p> <p>N:2:1 Declare associated pump or valve inoperable</p> <p>AND</p> <p>N:2:2 Comply with REQUIRED ACTION of 3.7.5 or 3.7.2 as applicable</p>	<p>48 hours <u>3.3-58</u></p> <p>Immediately</p> <p>Immediately</p>

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.3	NOT USED NOTE The continuity check may be excluded. Perform ACTUATION LOGIC TEST.	3.3-60 31 days on a STAGGERED TEST BASIS
SR 3.3.2.4	Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.5	Perform COT.	92 days
SR 3.3.2.6	Perform SLAVE RELAY TEST.	[92] da ys 18 months B-PS
(continued)		
SR 3.3.2.7	Not Used NOTE Verification of relay setpoints not required. Perform TADOT.	3.3-60 [92] days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.2.8 -----NOTE----- Verification of setpoint not required for manual initiation functions. ----- Perform TADOT.</p>	<p>18 months <u>B</u></p>
<p>SR 3.3.2.9 -----NOTE----- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. ----- Perform CHANNEL CALIBRATION.</p>	<p>18 months <u>B</u></p>
<p>SR 3.3.2.10 -----NOTE----- Not required to be performed for the turbine driven AFW pump until 24 hours after SG pressure is \geq [1000] 650 psig. ----- Verify ESFAS RESPONSE TIMES are within limit.</p>	<p><u>B</u> <u>B-PS</u> 18 months on a STAGGERED TEST BASIS</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.2.11 -----NOTE----- Verification of setpoint not required. ----- Perform TADOT.	<div style="text-align: right;"> <u>3.3-61</u> Once per reactor trip breaker cycle 18 months </div>

Table 3.3.2-1 (page 1 of 811)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT (a)	ED
1. Safety Injection							
a. Manual Initiation	1.2.3.4	2	B	SR 3.3.2.8	NA	NA	
b. Automatic Actuation Logic and Actuation Relays	1.2.3.4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA	
c. Containment Pressure - High \pm	1.2.3.4	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	\leq [3-86] 383 psig	\leq [3-6] 300 psig	<u>3.3-66</u> B-PS
d. Pressurizer Pressure - Low	1.2.3(b)	[3] 4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	\geq [1839] 1844 psig	\geq B	<u>B-PS</u> 1850 psig
e. Steam Line Pressure							
(1) Low	1.2. 3 (b)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	\geq [635] 594.6 (C) psig	\geq [675] 600 (C) psig	<u>B-PS</u> 3.3-01
(2) NOT USED High Differential Pressure Between Steam Lines	1.2.3	3 per steam line	D	[SR 3.3.2.1] SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	\leq [106] psig	\leq [97] psig	<u>3.3-01</u>
f. NOT USED High Steam Flow in Two Steam Lines	1.2.3 (d)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	(e)	(f)	
Coincident with T ₁ Low Low	1.2.3 (d)	1 per loop	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	\geq [550.6]°F	\geq [553]°F	

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT ^(a)

(continued)

- (a) ~~Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~ ED
- (b) ~~Above the P-11 (Pressurizer Pressure) interlock. Trip function may be blocked in this MODE below the P-11 (pressurizer interlock) setpoint.~~ 3.3-63
- B
- (c) ~~Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.~~ 3.3-105
- 3.3-01
- (e) ~~Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, and ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load.~~ 3.3-01
- 3.3-01
- (f) ~~Less than or equal to a function defined as ΔP corresponding to [40]% full steam flow between [0]% and [20]% load and then a ΔP increasing linearly from [40]% steam flow at [20]% load to [110]% full steam flow at [100]% load.~~ 3.3-01

Table 3.3.2-1 (page 23 of 811)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED		
						TRIP SETPOINT ^(a)		
1. Safety Injection (continued)							3.3-01	
g. High Steam Flow in Two Steam Lines	1.2.3^(d)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	(e)	(f)		
Coincident with Steam Line Pressure Low	1.2.3 ^(d)	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	> [635] ^(e) psig	> [675] psig		
2. Containment Spray							3.3-53	
a. Manual Initiation	1.2.3.4	2 per train with 2 trains coincident switches	B	SR 3.3.2.8	NA	NA		
b. Automatic Actuation Logic and Actuation Relays	1.2.3.4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA		
c. Containment Pressure							3.3-66	
High 3 (High High)	1.2.3 ^(d)	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ [12.31] psig	≤ [12.05] psig		
NOT USED High 3 (Two Loop Plants)							3.3-01	
	1.2.3	[3] sets of [2]	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ [12.31] psig	≤ [12.05] psig		

(continued)

- (a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.
- (c) Time constants used in the lead/lag controller are $t_1 >$ seconds and $t_2 <$ seconds
- (d) Above the P-12 (T_{low} - Low Low) interlock.
- (e) Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, and ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load.
- (f) Less than or equal to a function defined as ΔP corresponding to [40]% full steam flow between [0]% and [20]% load and then a ΔP increasing linearly from [40]% steam flow at [20]% load to [110]% full steam flow at [100]% load.

ED
B
3.3-01
3.3-01
3.3-01

Table 3.3.2-1 (page 34 of 81)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED	
						TRIP SETPOINT (a)	
3. Containment Isolation							
a. Phase A Isolation							
(1) Manual Initiation	1.2.3.4	2	B	SR 3.3.2.8	NA		NA
(2) Automatic Actuation Logic and Actuation Relays	1.2.3.4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA		NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.						
b. Phase B Isolation							
(1) Manual Initiation	1.2.3.4	2 per train (1) 2 trains coincident switches	B	SR 3.3.2.8	NA		3.3-53 NA
(2) Automatic Actuation Logic and Actuation Relays	1.2.3.4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA		NA
(3) Containment Pressure							B-PS 3.3-66
High-3 (High High)	1.2.3.4	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ [12.31] 22.3 psig		≤ [12.05] 22 psig
4. Steam Line Isolation							
a. Manual Initiation							
Manual Initiation	1.2 ⁽¹⁾ , 3 ⁽¹⁾	2 valve	F N	SR 3.3.2.8	NA		3.3-58 PS NA

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED	TRIP SETPOINT (a)
b. Automatic Actuation Logic and Actuation Relays	1.2 ⁽¹⁾ , 3 ⁽¹⁾	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA		NA

(continued)

- (a) ~~Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~
- (i) Except when all MSIVs are closed and ~~de-activated~~.

ED

B

Table 3.3.2-1 (page 45 of 81)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT (a)	ED
4. Steam Line Isolation (continued)							B-PS
						B	
						3.3-137	
c. Containment Pressure - High \geq High	1.2(i), 3(i)	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	\leq [6.61] 22.3 psig	\leq [6.35] 22.0 psig	
d. Steam Line Pressure							B-PS
(1) Low	1.2(i), 3(b)(i)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	\geq [635] 594.6 (c) psig	\geq [675] 600 (c) psig	
(2) Negative Rate - High	3(g)(i)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	\leq [121.6] 105.4 (h) psi/sec	\leq [110]100 (h) psi/sec	
e. NOT USED High Steam Flow in Two Steam Lines	1.2(i), 3(i)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	(e)	3.3-01	
Coincident with T₂₀ Low Low	1.2(i), 3(d)(i)	1 per loop	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	\geq [550.6]°F	\geq [553]°F	

(continued)

- (a) ~~Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~
- (b) ~~Above the P-11 (Pressurizer Pressure) interlock. Trip function may be blocked in this mode below the P-11 (pressurizer interlock) setpoint.~~
- (c) ~~Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds~~
- (d) ~~Above the P-12 (T₂₀ Low Low) interlock.~~
- (e) ~~Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load.~~
- (f) ~~Less than or equal to a function defined as ΔP~~

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3.3-63
B
3.3-105
3.3-01
3.3-01
3.3-01

~~corresponding to [40] full steam flow between [0] and [20] load and then a ΔP increasing linearly from [40] steam flow at [20] load to [110] full steam flow at [100] load.~~

(g) ~~Below the P-11 (Pressurizer Pressure) interlock. Trip function automatically blocked above P-11 (Pressurizer Pressure Interlock) setpoint and is automatically blocked below P-11 when Safety Injection on Steam Line Pressure-Low is not blocked.~~

3.3-63

(h) ~~Time constant utilized in the rate/lag controller is \leq [50] seconds. are $t_r = 50$ sec and $t_i = 50$ sec~~

3.3-105

(i) Except when all MSIVs are closed and ~~de-activated.~~

B

Table 3.3.2-1 (page 56 of 81)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT (a)	ED
4. Steam Line Isolation (continued)							3.3-01
f. High Steam Flow in Two Steam Lines	1.2(i) 3(i)	2 per steam line	0	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	(e)	(f)	
Coincident with Steam Line Pressure Low	1.2(i) 3(i)	1 per steam line	0	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	> [635] psig (e)	> [675] psig (f)	
							3.3-01
g. High Steam Flow	1.2(i) 3(i)	2 per steam line	0	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	< [25] % of full steam flow at no load steam pressure	< [] full steam flow at no load steam pressure	
Coincident with Safety Injection and	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.						
Coincident with T₁₂ Low Low	1.2(i) 3(d)(i)	[2] per loop	0	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	> [550.6] °F	> [553] °F	
							3.3-01
h. High High Steam Flow	1.2(i) 3(i)	2 per steam line	0	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	< [130] % of full steam flow at full load steam pressure	< [] of full steam flow at full load steam pressure	
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.						

(continued)

(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.
(d) Above the P-12 (T₁₂ Low Low) interlock.
(i) Except when all MSIVs are closed and [de-activated].

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3.3-01

Table 3.3.2-1 (page 67 of 811)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT (a)
5. Turbine Trip and Feedwater Isolation						
PS						
a. Automatic Actuation Logic and Actuation Relays	1.2(j) 1.2(j)	2 trains	H G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
B						
B-PS						
b. SG Water Level - High High (P-14)	1.2(j) 1.2(j)	3 per SG	I G	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 84.2 75.5%	≤ 92.4 75%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater						
3.3-58						
a. Manual	1.2.3	1 sw/dp	N	SR 3.3.2.8	NA	NA
a. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1.2.3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
3.3-01						
b. NOT USED	1.2.3	2 trains	G	SR 3.3.2.3	NA	NA
b. Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)	1.2.3	2 trains	G	SR 3.3.2.3	NA	NA
B-PS						
B						
3.3-46						
ed. SG Water Level - Low Low	1.2.3 1.2.3	3 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 30.4 6.8%	≥ 32.2 7.2%

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT (a)	ED
<u>Coincident with:</u>							<u>3.3-46</u>
1) RCS Loop ΔT Equivalent to Power > 50% RTP	1-2	4 (1/loop)	D	SR 3.3.2.5 SR 3.3.2.9	RCS Loop ΔT Variable Input > 51.5% RTP	RCS Loop ΔT variable input > 50% RTP	
With a time delay (TD)	1-2	4 (1/loop)	M	SR 3.3.2.5 SR 3.3.2.9	≤ (1.01) TD (j)	≤ TD (k)	
Or							
2) RCS Loop ΔT Equivalent to Power > 50% RTP	1-2	4 (1/loop)	D	SR 3.3.2.5 SR 3.3.2.9	RCS Loop ΔT variable Input > 51.5% RTP	RCS Loop ΔT Variable Input > 50% RTP	
With no time delay	1-2	4 (1/loop)	M	SR 3.3.2.5 SR 3.3.2.9	TD=0	TD=0	

(continued)

(a) ~~Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(j) Except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

(k) For Mode 3, the Trip Time Delay associated with the Steam Generator Water Level Low-Low channel must be less than or equal to 464.1 seconds.

(l) Steam Generator Water Level Low-Low Trip Time Delay

$$TD = B1(P) + B2(P) + B3(P) + B4$$

Where: P = RCS Loop ΔT Equivalent to Power (RTP), P = 50% RTP

TD = Time delay for Steam Generator Water Level Low-Low (in seconds)

$$B1 = -0.007128$$

$$B2 = +0.8099$$

$$B3 = -31.48$$

$$B4 = +464.1$$

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3.3-46

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Table 3.3.2-1 (page 79 of 81)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT (a)
6. Auxiliary Feedwater (continued)						
de. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
						<u>3.3-01</u>
ef. NOT USED Loss of Offsite Power	1-2-3	{3} per bus	F	SR 3.3.2.7 SR 3.3.2.9 SR 3.3.2.10	≥ [2912] V with < 0.8 sec time delay	≥ [2975] V with < 0.8 sec time delay
						<u>B-PS</u>
						<u>3.3-127</u>
fg. Undervoltage Reactor Coolant Pump	1-2	{3} % per bus	I	SR 3.3.2.7 SR 3.3.2.9 SR 3.3.2.10	≥ [69] % bus voltage 7730 volts	≥ [70] % bus voltage 8050 volts
						<u>3.3-116</u>
gh. NOT USED Trip of all Main Feedwater Pumps	1-2	{2} per pump	J	SR 3.3.2.8 SR 3.3.2.9 SR 3.3.2.10	≥ [] psig	≥ [] psig
						<u>3.3-01</u>
hi. NOT USED Auxiliary Feedwater Pump Suction Transfer on Suction Pressure Low	1-2-3	{2}	F	SR 3.3.2.1 SR 3.3.2.7 SR 3.3.2.9	≥ [20.53] [psia]	≥ [] [psia]
7. Automatic Switchover to Containment Sump						
a. Automatic Actuation Logic and Actuation Relays	1-2-3-4	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT ^(a)
b. Refueling Water Storage Tank (RWST) Level Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	$\geq [15]\%$ and $\leq []\%$	$\geq []$ and $\leq []$
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					

(continued)

~~(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

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Table 3.3.2-1 (page 811 of 811)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED
						TRIP SETPOINT ^(a)
7. NOT USED Automatic Switchover to Containment Sump (continued)						<u>3.3-01</u>
e. RWST Level Low	1.2.3.4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ [15]%	≥ [18]%
Coincident with Safety Injection and	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
Coincident with Containment Sump Level High	1.2.3.4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ [30] in. above el. [703] ft	≥ [] in. above el. [] ft
8. ESFAS Interlocks						
a. Reactor Trip. P-4	1.2.3	1 per train, 2 trains	F	SR 3.3.2.11	NA	NA
						<u>3.3-15</u>
						<u>3.3-44</u>
						<u>B-PS</u>
b. Pressurizer Pressure. P-11	1.2.3	3	L	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≤ [1996] [1920] psig	≤ [] psig
						<u>3.3-01</u>
c. NOT USED Low. P-12	1.2.3	[1] per loop	L	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≥ [550.6]°F	≥ [553]°F

(a) ~~Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

ED

3.3 INSTRUMENTATION

3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTIONS

NOTES-----

1. LCO 3.0.4 is not applicable.
2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable but at least one valid channel OPERABLE.	A.1 Restore required channel to OPERABLE status.	30 days <u>3.3-71</u>
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.8.	Immediately
C. -----NOTE----- Not applicable to hydrogen monitor channels. ----- One or more Functions with two required channels inoperable OPERABLE.	C.1 Restore one channel to OPERABLE status.	7 days <u>3.3-71</u>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION
<p>NOTE: Only applicable in MODES 1 and 2</p> <p>D. Two hydrogen monitor channels inoperable.</p>	D.1 Restore one hydrogen monitor channel to OPERABLE status.	72 hours <u>3.3-68</u>
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Enter the Condition referenced in Table 3.3.3-1 for the channel.	Immediately
F. As required by Required Action E.1 and referenced in Table 3.3.3-1.	F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.	6 hours 12 hours
G. As required by Required Action E.1 and referenced in Table 3.3.3-1.	G.1 Initiate action in accordance with Specification 5.6.8.	Immediately
H. As required by Required Action E.1 and referenced in Table 3.3.3-1.	H.1 Be in MODE 3	6 hours <u>3.3-68</u>

SURVEILLANCE REQUIREMENTS

-----NOTE-----
SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SURVEILLANCE	FREQUENCY
SR 3.3.3.1 Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days

SR 3.3.3.2

-----NOTE-----

1 Neutron detectors are excluded from CHANNEL CALIBRATION.

2 CHANNEL CALIBRATION for Containment Area Radiation may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source

3.3-20

Perform CHANNEL CALIBRATION.

18 months

B

Table 3.3.3-1 (page 1 of 2)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITION REFERENCED FROM REQUIRED ACTION E.1	
1. Power Range Neutron Flux (Wide Range NIS)	2	F	<u>3.3-71</u>
2. Source Range Neutron Flux	2 per steam generator	E	3.3-01
2. Steam Line Pressure		F	3.3-71
3. Reactor Coolant System (RCS) Hot Leg Temperature (Wide Range)	2 (1 per loop in two loops)	F	<u>3.3-71</u>
4. RCS Cold Leg Temperature (Wide Range)	2 (1 per loop in two loops)	F	<u>3.3-71</u>
5. RCS Pressure (Wide Range)	2	F	
6. Reactor Vessel Water Level Indication System	2	G	<u>3.3-71</u>
7. a) Containment Recirculation Sump Water Level (Wide Narrow Range)	2	F	<u>3.3-71</u>
b) Containment Reactor Cavity Sump Level (Wide Range)	2	E	<u>3.3-71</u>
8. a) Containment Pressure (Wide Range)	2	F	
b) Containment Pressure (Normal Range)	2	F	<u>3.3-71</u>
9. Containment Isolation Valve Position	2 per penetration flow path (a) (b)	F	
10. Containment Area Radiation (High Range)	2	G	
11. Hydrogen Monitors	2	F	<u>3.3-68</u>
12. Pressurizer Level	2	F	

13.	a) Steam Generator Water Level (Wide Range)	2 1 per steam generator	F	<u>3.3-71</u>
	b) Steam Generator Water Level (Narrow Range)	2 per steam generator	F	<u>3.3-71</u>
14.	Condensate Storage Tank Level	2	F	
15.	Core Exit Temperature In-core Thermocouples - Quadrant {1}	2 (c) per core quadrant	F	<u>3.3-71</u>
16.	Core Exit Temperature In-core Thermocouples - Quadrant {2}	2 (c) per core quadrant	F	<u>3.3-71</u>
17.	Core Exit Temperature In-core Thermocouples - Quadrant {3}	2 (c) per core quadrant	F	<u>3.3-71</u>
18.	Core Exit Temperature In-core Thermocouples - Quadrant {4}	2 (c) per core quadrant	F	<u>3.3-71</u>
19.	Auxiliary Feedwater Flow	2 1 per steam generator	F	<u>3.3-71</u>
20.	(New) Refueling Water Storage Tank Water Level	2	F	<u>3.3-71</u>

- (a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.
- (b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.
- (c) A channel consists of two ~~in-core exit~~ thermocouples (CETs).

Reviewer's Note: ~~Table 3.3.3-1 shall be amended for each unit as necessary to list:~~

~~(1) All Regulatory Guide 1.97, Type A instruments, and~~

~~(2) All Regulatory Guide 1.97, Category I, non Type A instruments in accordance with the unit's Regulatory Guide 1.97, Safety Evaluation Report.~~

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3.3 INSTRUMENTATION

3.3.4 Remote Shutdown System

LCO 3.3.4 The Remote Shutdown System Instrumentation Functions and the SD panel controls in Table 3.3.4-1 shall be OPERABLE.

3.3-94

APPLICABILITY: MODES 1, 2 and 3.

ACTIONS

-----NOTES-----

1. LCO 3.0.4 is not applicable.
 2. Separate Condition entry is allowed for each Function.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable.	A.1 Restore required Function to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	<p>-----NOTE----- Reactor Trip Breaker position is excluded from CHANNEL CHECK -----</p> <p>Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.</p>	<p><u>3.3-22</u> <u>B</u></p> <p>31 days <u>3.3-69</u></p>
SR 3.3.4.2	<p>Verify each required control circuit and transfer switch is capable of performing the intended function.</p>	<p>18 months <u>B</u></p>
SR 3.3.4.3	<p>-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION Reactor Trip Breaker position is excluded from CHANNEL CALIBRATION. -----</p> <p>Perform CHANNEL CALIBRATION for each required instrumentation channel.</p>	<p><u>3.3-84</u> <u>3.3-22</u></p> <p>18 months <u>B</u></p>
SR 3.3.4.4	<p>Perform TADOT of the reactor trip breaker open/closed indication.</p>	<p>18 months</p>

Table 3.3.4-1 (page 1 of 2)
Remote Shutdown System Instrumentation and Controls

3.3-128

NOTE

Reviewer's Note: This table is for illustration purposes only. It does not attempt to encompass every function used at every unit, but does contain the types of functions commonly found.

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FUNCTION/INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF FUNCTIONS
1. Reactivity Control	
a. Source Range Neutron Flux.	{1}
b. Reactor Trip Breaker Position	1 per trip breaker
	<u>B</u>
c. Manual Reactor Trip	{2}
2. Reactor Coolant System (RCS) Pressure Control	
a. Pressurizer Pressure of RCS Wide Range Pressure	1
b. Pressurizer Power Operated Relief Valve (PORV) Control and Block Valve Control	{1, controls must be for PORV & block valve on same line}
3. Decay Heat Removal via Steam Generators (SGs)	
a. RCS Hot Leg Temperature {loop 1 only}	1 per loop
b. RCS Cold Leg Temperature {loop 1 only}	1 per loop
c. AFW Controls Condensate Storage Tank Level	{1}2 of any 3 pumps
d. SG Pressure	1 per SG
e. Condensate Storage Tank Level	1
4. RCS Inventory Control	
a. Pressurizer Level	1
b. Charging Pump Controls	{1}2 of 2 pumps
c. Charging Flow	1
5. Safety Support Systems	
a. Emergency Diesel Generator Control	3 of 3 diesel generators
b. Component Cooling Water Control	any 2 of 3 pumps
c. Auxiliary Saltwater Control	2 of 2 pumps

3.3 INSTRUMENTATION

3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

LCO 3.3.5 ~~[[Three] One channels per bus of the loss of voltage DG start Function and two channels for initiation of load shed Function and [[three] two channels per bus of the degraded voltage Function with one timer per bus for DG start and initiation of load shed Function shall be OPERABLE.~~

3.3-133

APPLICABILITY: MODES 1, 2, 3, and 4.
When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more channels per bus inoperable.	A.1 -----NOTE----- The inoperable One channel may be bypassed for up to 4 2 hours for surveillance testing of other channels. ----- Place channel in trip. Declare the associated DG inoperable and enter the applicable Condition(s) and Required Action(s).	3.3-104 6 hours immediately
B. One or more Functions with two or more channels per bus inoperable.	B.1 Restore all but one channel to OPERABLE status.	1 hour 3.3-104

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C Required Action and associated Completion Time not met.	C.1 Enter applicable Condition(s) and Required Action(s) for the associated DG made inoperable by LOP DG start instrumentation.	Immed iate y <u>3.3-104</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
 SR 3.3.5.1 NOT USED Perform CHANNEL CHECK.	12 hours 
SR 3.3.5.2 Perform TADOT.	31 day 5 18 months <u>B-PS</u>

(continued)

3.3 INSTRUMENTATION

3.3.6 Containment Purge and Exhaust Isolation Instrumentation

LCO 3.3.6 The Containment Purge and Exhaust Isolation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: ~~MODES 1, 2, 3, and 4. According to Table 3.3.6-1~~
~~During CORE ALTERATIONS,~~
~~During movement of irradiated fuel assemblies within containment.~~

3.3-79

ACTIONS

-----NOTE-----
 Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NOTE: Only applicable in MODES 1, 2, 3, or 4. One radiation monitoring channel inoperable.	A.1 Restore the affected channel to OPERABLE status.	<div style="text-align: right;"><u>3.3-32</u></div> 4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. -----NOTE----- Only applicable in MODE 1, 2, 3, or 4. -----</p> <p>One or more Functions with one or more manual or automatic actuation trains inoperable.</p> <p><u>OR</u></p> <p>Two or more Both radiation monitoring channels inoperable.</p> <p><u>OR</u></p> <p>Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.</p>	<p>Immediately</p> <p style="text-align: center;"><u>3.3-32</u></p>

(continued)

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.6-1 to determine which SRs apply for each Containment Purge and Exhaust Isolation Function.

SURVEILLANCE	FREQUENCY
SR 3.3.6.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2 Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.6.3 Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.6.4 Perform GOT CFI .	<u>3.3-75</u> 92 days
SR 3.3.6.5 Perform SLAVE RELAY TEST.	[92] days <u>B-PS</u> 18 months
SR 3.3.6.6 NOT USED <div style="text-align: center;">NOTE Verification of setpoint is not required.</div> Perform TADOT.	<u>3.3-76</u> [18] months
SR 3.3.6.7 Perform CHANNEL CALIBRATION.	18 months <u>B</u>
SR 3.3.6.8 Verify ESF Containment Purge and Exhaust Isolation response time is within limits	18 months on a STAGGERED TEST BASIS <u>3.3-31</u>

Containment Purge and Exhaust Isolation Instrumentation
3.3.6

Table 3.3.6-1 (page 1 of 1)
Containment Purge and Exhaust Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT	<u>3.3-79</u>
1. NOT USED Manual Initiation		2	SR 3.3.6.6	NA	<u>3.3-77</u>
2. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4, 6, and (a)	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA	
3. Containment Radiation	1, 2, 3, 4 and (a)				
a. Gaseous 44 A/B	1, 2, 3, 4 (a)	(1) 2 1(b)	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	≤ [2 x background] Per ODCM	<u>3.3-32</u> <u>3.3-31</u>
b. Particulate		(1)	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	≤ [2 x background]	<u>3.3-32</u>
c. Iodine		(1)	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	≤ [2 x background]	<u>3.3-32</u>
d. Area Radiation		(1)	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7	≤ [2 x background]	<u>3.3-32</u>
4. Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a., for all initiation functions and requirements.				
(a) during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment					<u>3.3-31</u>
(b) only one monitor is required to be OPERABLE in MODE 6 or during movement of irradiated fuel assemblies within containment					<u>3.3-31</u>

3.3 INSTRUMENTATION

3.3.7 Control Room Emergency Filtration ~~Ventilation~~ System (GREFS ~~CRVS~~) Actuation Instrumentation PS

LCO 3.3.7 The GREFS ~~CRVS~~ actuation instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE. PS

APPLICABILITY: ~~MODES 1, 2, 3, 4, [5, and 6.] According to Table 3.3.7-1. During movement of irradiated fuel assemblies. [During CORE ALTERATIONS].~~

3.3-79

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	<p>A.1</p> <p style="text-align: center;">NOTE</p> <p style="text-align: center;">Place in toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable.</p> <p>Place one GREFS CRVS train in emergency [radiation protection] pressurization mode.</p>	<p>7 days <u>B-PS</u></p> <p><u>PS</u></p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. One or more Functions with two channels or two trains inoperable.</p>	<div style="border: 1px solid black; padding: 5px; margin-bottom: 10px;"> <p style="text-align: center;"><u>NOTE</u></p> <p style="text-align: center;">Place in the toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable.</p> </div> <p>B.1.1 Place one GREFSCRVS train in emergency [radiation protection] pressurization mode.</p> <p style="text-align: center;"><u>AND</u></p> <p>B.1.2 Enter applicable Conditions and Required Actions for one GREFSCRVS train made inoperable by inoperable GREFSCRVS actuation instrumentation.</p> <p style="text-align: center;"><u>OR</u></p> <p>B.2 Place both trains in emergency [radiation protection] mode.</p>	<p style="text-align: right;"><u>PS</u></p> <p>Immediately</p> <p style="text-align: right;"><u>B-PS</u></p> <p>Immediately</p> <p style="text-align: right;"><u>PS</u></p> <p style="text-align: right;"><u>3.3-51</u></p> <p>Immediately</p>
<p>C. Required Action and associated Completion Time for Condition A or B not met in MODE 1, 2, 3, or 4.</p>	<p>C.1 Be in MODE 3.</p> <p style="text-align: center;"><u>AND</u></p> <p>C.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time for Condition A or B not met during movement of irradiated fuel assemblies [or during CORE ALTERATIONS] .	D.1 Suspend CORE ALTERATIONS	Immediately <u>B</u> <u>3.3-118</u>
	AND D.2 Suspend movement of irradiated fuel assemblies.	Immediately <u>B</u>
E. Required Action and associated Completion Time for Condition A or B not met in MODE 5 or 6.	E.1 Initiate action to restore one GREFS CRVS train to OPERABLE status.	Immediately <u>B-PS</u> <u>B</u>

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.7-1 to determine which SRs apply for each GREFS ~~CRVS~~ B-PS
Actuation Function.

SURVEILLANCE	FREQUENCY
SR 3.3.7.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2 Perform COT CFT .	92 days <u>3.3-75</u>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.3	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.7.4	Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.7.5	Perform SLAVE RELAY TEST.	92 days 18 months <u>B-PS</u>
SR 3.3.7.6	-----NOTE----- Verification of setpoint is not required. ----- - Perform TADOT.	<u>B</u> 18 months
SR 3.3.7.7	Perform CHANNEL CALIBRATION.	18 months <u>B</u>

Table 3.3.7-1 (page 1 of 1)
GREFS ~~CRVS~~ Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT	<u>3.3-79</u>
1. Manual Initiation	1, 2, 3, 4, 5, 6, and (a)	2 trains	SR 3.3.7.6	NA	
2. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4, 5, 6, and (a)	2 trains	SR 3.3.7.3 SR 3.3.7.4 SR 3.3.7.5	NA	
3. Control Room Radiation	1, 2, 3, 4, 5, 6, and (a)				
a. Control Room Atmosphere Air Intakes	1, 2, 3, 4, 5, 6, and (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.7	≤ 2 mR/hr	<u>3.3-102</u> <u>B</u>
b. Control Room Air Intakes		[2]	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.7	≤ [2] mR/hr	<u>3.3-102</u>
4. Safety Injection			Refer to LCO 3.3.2. "ESFAS Instrumentation." Function 1, for all initiation functions and requirements.		

~~(a) During movement of irradiated fuel assemblies~~

3.3-79

3.3 INSTRUMENTATION

3.3.8 Fuel Building Air-Cleanup ~~Ventilation~~ System (FBAGS ~~FBVS~~) Actuation Instrumentation.

PS

LCO 3.3.8 The ~~FBAGS~~~~FBVS~~ actuation instrumentation for each Function in Table 3.3.8-1 shall be OPERABLE.

PS

APPLICABILITY: According to Table 3.3.8-1.

ACTIONS

-----NOTE-----

1. ~~Separate Condition entry is allowed for each Function.~~

2. ~~LCO 3.0.3 is not applicable~~

3.3-34

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more Functions with one or more channels or trains inoperable.</p>	<p>A.1.1 Place one FBACS train in operation. Restore the inoperable monitors to OPERABLE status.</p> <p><u>AND</u></p> <p>A.1.2.1 Install an appropriate portable continuous monitor with the same alarm setpoint.</p> <p><u>OR</u></p> <p>A.1.2.2 Station an individual qualified in radiation protection procedures with a dose rate monitoring device in the spent fuel pool area.</p> <p><u>OR</u></p> <p>A.1.2.3.1 Place one FBVS train in the Iodine Removal mode.</p> <p><u>AND</u></p> <p>A.1.2.3.2 Enter applicable Conditions and Required Actions of LCO 3.7.13, "Fuel Building Air Cleanup System (FBACS)" for one train made inoperable by inoperable actuation instrumentation.</p>	<p>730 days <u>3.3-82</u></p> <p><u>Immediately</u></p> <p><u>Immediately</u></p> <p><u>Immediately</u></p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. NOT USED One or more Functions with two channels or two trains inoperable.</p> <p>B. (continued)</p>	<p>B.1.1 Place one FBACS train in operation</p> <p>AND</p> <p>B.1.2 Enter applicable Conditions and Required Actions of LCO 3.7.13, "Fuel Building Air Cleanup System (FBACS)," for one train made inoperable by inoperable actuation instrumentation.</p> <p>OR</p> <p>B.2 Place both trains in emergency [radiation protection] mode.</p>	<p>Immediately</p> <p>3.3-82</p> <p>Immediately</p> <p>(continued)</p> <p>Immediately</p>
<p>C. Required Action and associated Completion Time for Condition A or B not met during movement of irradiated fuel assemblies in the fuel building.</p>	<p>C.1 Suspend movement of irradiated fuel assemblies in the fuel building.</p>	<p>Immediately</p> <p><u>3.3-82</u></p>
<p>D. Required Action and associated Completion Time for Condition A or B not met in MODE 1, 2, 3, or 4.</p>	<p>D.1 Be in MODE 3.</p> <p>AND</p> <p>D.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.8-1 to determine which SRs apply for each FBACS ~~FBVS~~ Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.8.2	Perform GOT CFT .	92 day s <u>3.3-75</u>
(continued)		
[SR 3.3.8.3 NOT USED	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS]
SR 3.3.8.4	-----NOTE----- Verification of setpoint is not required. ----- Perform TADOT.	18 months <u>B</u>
SR 3.3.8.5	Perform CHANNEL CALIBRATION.	18 months <u>B</u>

Table 3.3.8-1 (page 1 of 1)
FBAGS Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
				<u>3.3-82</u>
1. Manual Initiation	[1.2.3.4] (a)	2 2	SR 3.3.8.4 SR 3.3.8.4	NA NA
2. Automatic Actuation Logic and Actuation Relays	1.2.3.4 (a)	2 trains	SR 3.3.8.3	NA
3. Fuel Building Radiation				<u>3.3-82</u>
				<u>B-PS</u>
a. Gaseous Spent Fuel Pool	[1.2.3.4] (a)	[2]	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.5	< [2] 75 mR/hr
				<u>B-PS</u>
b. Particulate New Fuel Storage Vault	[1.2.3.4] (a)	[2]	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.5	< [2] 15 mR/hr
c. Gaseous	(a)	2	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.5	per ODCM

(a) During movement of irradiated fuel assemblies in the fuel building.

~~(b) The requirements for FHBV mode change will not be applicable to the spent fuel storage pool or new fuel storage vault monitors following the installation of the gaseous monitors RM-45A and 45B.~~

3.3-82

~~(c) The requirements for FHBV mode change are applicable following the installation of RM-45A and 45B.~~

3.3-82

3.3 INSTRUMENTATION

3.3.9 Boron Dilution Protection System (BDPS)

~~NOT USED~~

~~LCO 3.3.9 Two trains of the BDPS shall be OPERABLE.~~

~~APPLICABILITY: MODES [2,] 3, 4, and 5.~~

~~NOTE~~

~~The boron dilution flux doubling signal may be blocked in MODES 2 and 3 during reactor startup.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One train inoperable.	A.1 Restore train to OPERABLE status.	72 hours
B. Two trains inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A not met.	B.1 Suspend operations involving positive reactivity additions. <u>AND</u> B.2.1 Restore one train to OPERABLE status. <u>OR</u> B.2.2.1 Close unborated water source isolation valves. <u>AND</u>	Immediately 1 hour 1 hour (continued)
B. (continued)	B.2.2.2 Perform SR 3.1.1.1.	1 hour <u>AND</u> Once per 12 hours thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.9.1 Perform COT.	[92] days
SR 3.3.9.2 Perform CHANNEL CALIBRATION.	[18] months

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes- Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is

**Methodology For Mark-up of NUREG-1431 Specifications
(Continued)**

"struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG.-1431 BASES

MARK-UP OF NUREG.-1431 BASES CONTENTS

Mark-up

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B 3.3.7.....	B 3.3-172
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B 3.3.9.....	N/A

Methodology

2 pages

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (A00s) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this specification as the Trip Setpoints, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During A00s, which are those events expected to occur ~~one or more times~~ more than once during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);
2. Fuel centerline melt shall not occur; and
3. The RCS pressure SL of ~~2750 psia~~ 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during A00s.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a

(continued)

BASES

BACKGROUND
(continued)

different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RTS instrumentation is segmented into four distinct but interconnected modules as illustrated in Figure [], FSAR, Chapter [7] (Ref. 1), and as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;
2. Signal Process Control and Protection System, including Analog/Digital Protection System, Nuclear Instrumentation System (NIS), field contacts, and protection channel sets: provides signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications;
3. Solid State Protection System (SSPS), including input, logic, and output bays: initiates proper unit shutdown and/or ESF actuation in accordance with the defined logic, which is based on the bistable outputs from the signal process control and protection system; and
4. Reactor trip switchgear, including reactor trip breakers (RTBs) and bypass breakers: provides the means to interrupt power to the control rod drive mechanisms (CRDMs) and allows the rod cluster control assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the Trip Setpoint and Allowable

(continued)

BASES

BACKGROUND

Field Transmitters or Sensors (continued)

Values. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

Signal Process Control and Protection System

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in the FSAR (References 1, 2, 3, 9, 10, & 11) Chapter [7] (Ref. 1), Chapter [6] (Ref. 2), and Chapter [15] (Ref. 3). If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the SSPS for decision evaluation, except in the case of the seismic turbine stop valve position, auto stop oil pressure, 12 kV bus and RCP breaker inputs which do not go through signal conditioning. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, the main control board, the unit computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails, such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. In the case of the Digital Feedwater Control System (DFWCS), the median/signal select (MSS) feature prevents control/protection interaction even though there are only three inputs and 2-out-of-3 logic. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor

(continued)

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BACKGROUND

Signal Process Control and Protection System (continued)

prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 1.

Two logic channels are required to ensure no single random failure of a logic channel will disable the RTS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing trip.

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., \pm rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in Reference 1. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the "RTS/ESFAS Setpoint Methodology WCAP-11082 Rev. 2, Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Station - Egel 21 Version, May 1993 (Ref. 6). The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

(continued)

BASES

BACKGROUND
(continued)

Trip Setpoints and Allowable Values

Setpoints in accordance with the Allowable Value ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed). Note that in the accompanying LCO 3.3.1, the Trip Setpoints of Table 3.3.1-1 are the LSSS.

Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements of Reference 2. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal or in the case of the Power Range channels the test signal is added to the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

The Trip Setpoints and Allowable Values listed in Table 3.3.1-1 are based on the methodology described in Reference 6, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip and/or ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

(continued)

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BACKGROUND

Solid State Protection System (continued)

The SSPS performs the decision logic for actuating a reactor trip or ESF actuation, generates the electrical output signal that will initiate the required trip or actuation, and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable outputs from the signal processing equipment or relay contact input (RCP breaker, 12KV UV/UF, seismic, etc.) are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various unit upset and accident transients. If a required logic matrix combination is completed, the system will initiate a reactor trip or send actuation signals via master and slave relays to those components whose aggregate function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Reactor Trip Switchgear

The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity. Each RTB is equipped with a bypass breaker to allow testing of the RTB while the unit is at power. During normal operation the output from the SSPS is a voltage signal that energizes the undervoltage coils in the RTBs and bypass breakers, if in use. When the required logic matrix combination is completed, the SSPS output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTBs and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each reactor trip breaker is also equipped with an automatic shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the SSPS. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

The decision logic matrix Functions are described in the functional diagrams included in Reference 2. In addition to

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Reactor Trip Switchgear (continued)

the reactor trip or ESF, these diagrams also describe the various "permissive interlocks" that are associated with unit conditions. Each train has a built in testing device that can automatically test the decision logic matrix Functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

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The RTS functions to maintain the SLs applicable limits during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis described in Reference 3 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the accident analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RTS trip Functions that were credited in the accident analysis.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Generally four OPERABLE instrumentation channels in a two-out-of-four configuration are required when one RTS channel is also used as a control system input. In the case of the DFWCS, the MSS feature prevents control/protection interaction even though there are only three inputs and a 2-out-of-3 logic. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In

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this case, the RTS will still provide protection, even with random failure of one of the other three protection channels. Three operable instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RTS trip and disable one RTS channel. The two-out-of-three and two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint.

The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel is controlled by a manual reactor trip switch. Each channel activates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE ~~(1 out of 2 coincidence)~~. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if the shutdown rods or control rods are withdrawn or the Control Rod Drive (CRD) System is capable of withdrawing the shutdown rods or the control rods. In this condition, inadvertent control rod withdrawal is

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1. Manual Reactor Trip (continued)

possible. In MODE 3, 4, or 5, manual initiation of a reactor trip does not have to be OPERABLE if the CRD System is not capable of withdrawing the shutdown rods or control rods. If the rods cannot be withdrawn from the core, there is no need to be able to trip the reactor because all of the rods are inserted. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System and the ~~Steam Generator (SG) Water Level Control System~~. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

a. Power Range Neutron Flux-High

The Power Range Neutron Flux-High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB ~~during power operations. These can be caused by rod withdrawal or reductions in RCS temperature, fuel damage.~~ Reactivity excursions can be caused by rod withdrawal or inadvertent CVCS malfunction, or for example, by sudden changes in RCS coolant temperature such as a feedwater system malfunction (Ref. 12).

The LCO requires all four of the Power Range Neutron Flux-High channels to be OPERABLE (2-out-of-4 coincidence).

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux-High trip must be OPERABLE. This Function

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a. Power Range Neutron Flux-High (continued)

will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect neutron levels in this range. In these MODES, the Power Range Neutron Flux-High does not have to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

b. Power Range Neutron Flux-Low

The LCO requirement for the Power Range Neutron Flux-Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux-Low channels to be OPERABLE (2 out of 4 coincidence).

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than approximately or equal to 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection

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b. Power Range Neutron Flux-Low (continued)

against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Power Range Neutron Flux Rate

The Power Range Neutron Flux Rate trips use the same channels as discussed for Function 2 above.

a. Power Range Neutron Flux-High Positive Rate

The Power Range Neutron Flux-High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function complements the Power Range Neutron Flux-High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

The LCO requires all four of the Power Range Neutron Flux-High Positive Rate channels to be OPERABLE (~~2-out~~
~~of 4 coincidence~~).

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux-High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

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b. Power Range Neutron Flux-High Negative Rate

The Power Range Neutron Flux-High Negative Rate trip Function ensures that protection is provided for multiple rod drop accidents. At high power levels, a multiple rod drop accident could cause local flux peaking that would result in an unconservative local DNBR. DNBR is defined as the ratio of the heat flux required to cause a DNB at a particular location in the core to the local heat flux. The DNBR is indicative of the margin to DNB. No credit is taken for the operation of this Function for those rod drop accidents in which the local DNBRs will be greater than the limit.

The LCO requires all four Power Range Neutron Flux-High Negative Rate channels to be OPERABLE (~~2-out-of-4 coincidence~~).

In MODE 1 or 2, when there is potential for a multiple rod drop accident to occur, the Power Range Neutron Flux-High Negative Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-High Negative Rate trip Function does not have to be OPERABLE because the core is not critical and DNB is not a concern. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the required SDM is increased during refueling operations. In addition, the NIS power range detectors cannot detect neutron levels present in this MODE.

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux-Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY 4. Intermediate Range Neutron Flux (continued)

do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE (1-out-of-2 coincidence). Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2 above the P-6 setpoint, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux-High Setpoint trip and the Power Range Neutron Flux-High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range detectors cannot detect neutron levels present in this MODE.

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to

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5. Source Range Neutron Flux (continued)

the Power Range Neutron Flux-Low Setpoint and ~~Intermediate Range Neutron Flux trip Functions~~. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 2 below P-6, 3, 4, and 5 with the Rod Control System capable of rod withdrawal or all rods not fully inserted. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. The LCO also requires one channel of the Source Range Neutron Flux to be OPERABLE in MODE 3, 4, or 5 with RTBs open or the control rods incapable of withdrawal. In this case, the source range Function is to provide control room indication and input to the Boron Dilution Protection System (BDPS). The outputs of the Function to RTS logic are not required OPERABLE in MODE 6 or when the RTBs are open or all rods are fully inserted and the Rod Control System is incapable of withdrawal.

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The Function also provides visual neutron flux indication in the control room.

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE (1-out-of-2 coincidence). Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux-Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range neutron flux trip may be manually blocked and the high voltage to the detectors may be de-energized; detectors are de-energized and inoperable. Below the P-6 setpoint, the source range neutron flux trip is automatically reinstated and the high voltage to the detectors is automatically energized.

In MODE 3, 4, or 5 with the reactor shut down, but with the Rod Control System capable of rod withdrawal or all rods not fully inserted, the Source Range Neutron Flux trip Function must also be OPERABLE (1-out-of-2 coincidence). If the CRD System is capable of rod withdrawal, the Source Range Neutron Flux trip must be

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5. Source Range Neutron Flux (continued)

~~OPERABLE~~ to provide core protection against a rod withdrawal accident. If the ~~CRD Rod Control~~ System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like an ~~uncontrolled~~ boron dilution. ~~These inputs are provided to the BDPS.~~ The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation."

6. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection and ~~it protects against vessel exit bulk boiling and ensures that the exit quality is within the limits defined by the DNBR correlation.~~ The inputs to the Overtemperature ΔT trip include ~~a~~ pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. ~~The Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase.~~ The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure—the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution— $f(\Delta I)$, the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the
- NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.
- Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

ΔT_0 , as used in the overtemperature and overpower ΔT trips, represents the 100 percent RTP value of ΔT as measured by the

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plant for each loop. For the initial startup of a refueled core, ΔT_0 is initially assumed to be the same as the last measured ΔT value from the previous cycle until ΔT is measured again at full power. Accurate determination of the loop specific ΔT values should be made quarterly when performing the incore/excore recalibration at steady-state conditions (i.e., power distributions not affected by xenon or other transient conditions). The variation in indicated ΔT between loops is due to the variance in both real hot leg temperatures and hot leg temperature measurement biases. The real hot leg temperature variance between loops is primarily caused by asymmetrical flow in the upper plenum, and the difference in hot leg temperature measurement bases, primarily caused by differences in hot leg temperature streaming error between loops. The change in the indicated loop ΔT s with burn up is caused primarily by the change in the hot leg streaming biases as the radial power distribution changes.

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two loops. ~~At some units,~~ The pressure and temperature signals are used for other control functions. ~~For these units, thus~~ the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE ~~for two and four loop units (the LCO requires all three channels on the Overtemperature ΔT trip Function to be OPERABLE for three loop units)~~. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB ~~(2-out-of-4 coincidence)~~. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

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7.

Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions for Condition I and II event (Ref. 12). This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature—including dynamic compensation for the delays between the core and the temperature measurement system.

ΔT_0 , as used in the overtemperature and overpower ΔT trips, represents the 100 percent RTP value of ΔT as measured by the plant for each loop. For the initial startup of a refueled core, ΔT_0 is initially assumed to be the same as the last measured ΔT value from the previous cycle until ΔT is measured again at full power. Accurate determination of the loop specific ΔT values should be made quarterly when performing the incore/excore recalibration at steady-state conditions (i.e., power distributions not affected by xenon or other transient conditions). The variation in indicated ΔT between loops is due to the variance in both real hot leg temperatures and hot leg temperature measurement biases. The real hot leg temperature variance between loops is primarily caused by asymmetrical flow in the upper plenum, and the difference in hot leg temperature measurement biases is primarily caused by differences in hot leg temperature streaming error between loops. The change in the indicated loop ΔT s with burn up is caused primarily by the change in the hot leg streaming biases as the radial power distribution changes.

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops. ~~At some units, the temperature signals are used for other control functions. At these units, thus,~~ the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to

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generate a turbine runback prior to reaching the Allowable
Value ~~trip setpoint~~. A turbine runback will reduce turbine

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power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels ~~for two and four loop units (three channels for three loop units)~~ of the Overpower ΔT trip Function to be OPERABLE (2-out-of-4 coincidence). Note that the Overpower ΔT trip Function receives input from ~~consequences of small steamline breaks, as reported in WCAP 9227, Ref. 11, and steamline breaks with coincident control rod withdrawal (Ref. 12).~~

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7. Overpower ΔT (continued)

channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure-High and -Low trips and the Overtemperature ΔT trip. ~~At some units, the Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System. For these units thus, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.~~

a. Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

The LCO requires four channels ~~for two and four loop units (three channels for three loop units)~~ of Pressurizer Pressure-Low to be OPERABLE (2-out-of-4 coincidence).

In MODE 1, when DNB is a major concern, the Pressurizer Pressure-Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 ~~Low Pressure Permissive~~ interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent

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a. Pressurizer Pressure - Low (continued)

(P-13)). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, ~~no conceivable power distributions can occur that would cause DNB concerns. There is insufficient heat production to be concerned about DNB~~

b. Pressurizer Pressure - High

The Pressurizer Pressure - High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

The LCO requires four channels ~~for two and four loop units (three channels for three loop units)~~ of the Pressurizer Pressure - High to be OPERABLE ~~(2-out-of-4 coincidence)~~.

The Pressurizer Pressure - High LSSS is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

In MODE 1 or 2, the Pressurizer Pressure - High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure - High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will ~~usually~~ be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. ~~Additionally, two low temperature overpressure protection systems channels provide overpressure protection with the PORVs when below MODE-4 the low temperature cut-off specified in the pressure and temperature limits report (PTLR).~~

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9. Pressurizer Water Level-High

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level-High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level-High trip must be OPERABLE (2-out-of-3 coincidence). This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

10. Reactor Coolant Flow-Low

a. ~~Reactor Coolant Flow Low (Single Loop)~~

~~The Reactor Coolant Flow Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately 48 RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.~~

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~~a. Reactor Coolant Flow Low (Single Loop)
(continued)~~

~~The LCO requires three Reactor Coolant Flow Low channels per loop to be OPERABLE in MODE 1 above P 8. In MODE 1 above the P 8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core. In MODE 1 below the P 8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip (Function 10.b) because of the lower power level and the greater margin to the design limit DNBR.~~

~~b. Reactor Coolant Flow Low (Two Loops)~~

The Reactor Coolant Flow-Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

~~Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.~~

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE (2 out of 3 coincidence in one loop).

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow-Low (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

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11. Reactor Coolant Pump (RCP) Breaker Position

~~Both RCP Breaker Position trip Functions operate together on two sets of auxiliary contacts, with one set on each RCP breaker. These Functions anticipate the Reactor Coolant Flow Low trips to avoid RCS heatup that would occur before the low flow trip actuates.~~

a. Reactor Coolant Pump Breaker Position (Single Loop)
~~NOT USED~~

~~The RCP Breaker Position (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open above the P-8 setpoint, a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow Low (Single Loop) Trip Setpoint is reached.~~

~~The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE. One OPERABLE channel is sufficient for this trip Function because the RCS Flow Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.~~

~~This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.~~

~~In MODE 1 above the P-8 setpoint, when a loss of flow in any RCS loop could result in DNB conditions in the core, the RCP Breaker Position (Single Loop) trip must be OPERABLE. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR.~~

(continued)

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b. Reactor Coolant Pump Breaker Position (Two Loops)

The RCP Breaker Position (~~Two Loops~~) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The position of each RCP breaker is monitored. Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two or more loops will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (~~Two Loops~~) Trip Setpoint is reached.

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE (~~2-out-of-4 coincidence~~). One OPERABLE channel is sufficient for this Function because the RCS Flow-Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the RCP Breaker Position (~~Two Loops~~) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in two RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

(continued)

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12. Undervoltage Reactor Coolant Pumps

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP bus is monitored by two relays each. Above the P-7 setpoint, a loss of voltage detected on two or more RCP buses, i.e. a complete loss of flow event will initiate a reactor trip. For this event, the under voltage trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires three Undervoltage RCPs channels (one per phase) per bus to be OPERABLE (1 per bus both buses).

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level, since there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in two or more of four RCS loops is automatically enabled. This Function uses the same relays as the ESFAS Function 6.f. "Undervoltage Reactor Coolant Pump (RCP)" start of the auxiliary feedwater (AFW) pumps.

13. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper An adequate coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 setpoint, a loss of frequency detected on two or more by two relays on one RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

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13. Underfrequency Reactor Coolant Pumps (continued)

The LCO requires three two Underfrequency RCPs channels per bus to be OPERABLE (1-per-bus both busses).

In MODE 1 above the P-7 setpoint, the Underfrequency RCPs trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level since there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

14. Steam Generator Water Level - Low Low

The SG Water Level - Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes in the event of a loss of feedwater flow to one or more SGs. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This Function also performs the ESFAS function of starting the AFW pumps on low low SG level.

The LCO requires four three channels of SG Water Level - Low Low per SG (1-per-SG in one SG) and four channels of RCS ΔT (1/loop) to be OPERABLE. The installation of the median signal selector (MSS) and four channels of RCS ΔT (1/loop) effectively eliminates the possibility that a single random failure could cause a control system action that results in a condition requiring protection action, and also prevent proper operation of a protection system channel designed to protect against the condition. Thus, the MSS prevents interaction between the feedwater control and reactor protection systems in accordance with the requirements of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." Removal of this interaction eliminates the need for the low feedwater flow reactor trip. The MSS will functionally separate steam generator narrow range level protection channels (low-low steam generator water level trip) to provide compliance with IEEE 279-1971 and satisfy the original design basis for four loop units in which these channels are shared between protection and control. In two, three, and four loop units where three SG Water Levels are dedicated to the RTS, only three channels per SG are required to be OPERABLE. This trip is actuated on two out of three low-low water level signals occurring in any steam generator. If a low-low water level condition is detected in one steam generator, signals shall be generated to trip the reactor and start the motor

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driven auxiliary feedwater pumps. If a low-low water level condition is detected in two or more steam generators, a signal is generated to start the turbine driven auxiliary feedwater pump as well.

The signals to actuate reactor trip and start auxiliary feedwater pumps may be delayed through the use of a Trip Time Delay (TTD) system for reactor power levels below 50% of RTP. Low-low water level in any protection set in any steam generator will generate a signal which starts an elapsed time trip delay timer. The allowable trip time delay is based upon the prevailing power level at the time the low-low level trip setpoint is reached. If power level rises after the trip time delay setpoints have been determined, the trip time delay is re-determined (i.e., decreased) according to the increase in power level. However, the trip time delay is not changed if the power level decreases after the delay has been determined. The use of this delay allows added time for natural steam generator level stabilization or operator intervention to avoid an inadvertent protection system actuation.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level - Low Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to

(continued)

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14. Steam Generator Water Level-Low Low (continued)

ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level-Low Low Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System in MODE 3 (and 4 prior to going on RHR) and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

15. Steam Generator Water Level-Low, Coincident With Steam Flow/Feedwater Flow Mismatch

~~NOT USED~~

~~SG Water Level Low, in conjunction with the Steam Flow/Feedwater Flow Mismatch, ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes. In addition to a decreasing water level in the SG, the difference between feedwater flow and steam flow is evaluated to determine if feedwater flow is significantly less than steam flow. With less feedwater flow than steam flow, SG level will decrease at a rate dependent upon the magnitude of the difference in flow rates. There are two SG level channels and two Steam Flow/Feedwater Flow Mismatch channels per SG. One narrow range level channel sensing a low level coincident with one Steam Flow/Feedwater Flow Mismatch channel sensing flow mismatch (steam flow greater than feed flow) will actuate a reactor trip.~~

~~The LCO requires two channels of SG Water Level Low coincident with Steam Flow/Feedwater Flow Mismatch.~~

~~In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level Low coincident with Steam Flow/Feedwater Flow Mismatch trip must be OPERABLE. The normal source of water for the SGs is the MFW System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5,~~

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15. Steam Generator Water Level-Low, Coincident With Steam Flow/Feedwater Flow Mismatch (continued)

~~or 6, the SG Water Level-Low coincident with Steam Flow/Feedwater Flow Mismatch Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System in MODE 3 and by the RHR System in MODE 4, 5, or 6. The MFW System is in operation only in MODE 1 or 2 and, therefore, this trip Function need only be OPERABLE in these MODES.~~

16. Turbine Trip

a. Turbine Trip-Low Fluid Auto Stop Oil Pressure

The Turbine Trip-Low Fluid Auto Stop Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately less than or equal to 50% power, will not actuate a reactor trip. Three pressure switches monitor the control trip oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three channels of Turbine Trip-Low Fluid Autostop Oil Pressure to be OPERABLE in MODE 1 above P-9 (2-out-of-3 coincidence).

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip.

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a. Turbine Trip-Low Fluid Auto Stop Oil Pressure (continued)

and the Turbine Trip-Low Fluid Auto Stop Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip-Turbine Stop Valve Closure

The Turbine Trip-Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip ~~from a power level below the P-9 setpoint, approximately 50% power.~~

~~This action will not actuate a reactor trip.~~ The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close.

Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. ~~Any turbine trip from a power level below the P-9 setpoint, less than or equal to a maximum setpoint of 50 percent power, will not actuate a reactor trip.~~ This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip-Low Fluid Auto Stop Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated ~~above P-9.~~

The LSSS for this Function is set to assure channel trip occurs when the associated stop valve is completely closed.

The LCO requires four Turbine Trip-Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. All four channels must trip to cause reactor trip.

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump ~~and Reactor Control Systems~~. In MODE 2, 3, 4, 5, or 6, there is no potential for

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b. Turbine Trip - Turbine Stop Valve Closure
(continued)

a load rejection, and the Turbine Trip - Stop Valve Closure trip Function does not need to be OPERABLE.

17. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for the small break LOCA but rod insertion is not credited for the large break LOCA (Ref. 3). However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoint and Allowable Values are not applicable to this Function. The SI Input is provided by relay logic in the SSPS circuitry of ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2 (1-out-of-2 coincidence).

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

18. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE

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18. Reactor Trip System Interlocks (continued)

when the associated reactor trip functions are outside the applicable MODES. These are:

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range Neutron Flux reactor trip and allows the high voltage to be de-energized. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range, and when the source range trip is blocked, the high voltage to the detectors is also removed;
- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip; and
- on increasing power, the P-6 interlock provides a backup block signal to the source range flux doubling circuit. Normally, this function is manually blocked by the control room operator during the reactor startup.

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint (1-out-of-2 coincidence).

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this function will no longer be necessary.

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a. Intermediate Range Neutron Flux, P-6 (continued)

In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection.

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Pressure, P-13 interlock. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure-Low;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (Two Loops);
- RCPs Breaker Open (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure-Low;

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b. Low Power Reactor Trips Block, P-7 (continued)

- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

Trip Setpoint and Allowable Value are not applicable to the P-7 interlock because it is a logic Function and thus has no parameter with which to associate an LSSS. The P-7 train is operable if the P-10 and P-13 interlocks are in their required states based on plant conditions.

The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1 (1-out-of-2 coincidence).

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8

The Power Range Neutron Flux, P-8 interlock is actuated at approximately 48-85% power as determined by two-out-of-four NIS power range detectors. The P-8 interlock automatically enables the Reactor Coolant Flow - Low (Single Loop) and RCP Breaker Position (Single Loop) reactor trips on low flow in one or more RCS loops on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately 48-85% power. On decreasing

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c. Power Range Neutron Flux, P-8 (continued)

power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated at approximately less than or equal to 50% power as determined by two-out-of-four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip-Low Fluid Auto Stop Oil Pressure and Turbine Trip-Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will may challenge the pressurizer PORVs due to the mismatch between reactor power and cause a load rejection beyond the capacity capacities of the Steam Dump and Reactor Control Systems. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

The LCO requires four~~three~~ channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1 (~~2-out-of-3 coincidence~~).

~~In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a significant load rejection beyond the capacity capacities of the Steam Dump and Reactor Control Systems.~~

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e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal;
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux-Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a back up signal to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detectors high voltage and allows manual block of the IR rod stop;
- the P-10 interlock provides one of the two inputs to the P-7 interlock; and
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux-Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).
- on decreasing power, the P-10 interlock automatically defeats the block of the source range neutron flux trip and with P-6 energizes the source range high voltage.

The LCO requires ~~four~~ ^{three} channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2 (~~2-out of 3~~).

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a

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e. Power Range Neutron Flux, P-10 (continued)

startup or shutdown by the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

f. Turbine Impulse Chamber Pressure, P-13

The Turbine Impulse Chamber Pressure, P-13 interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than approximately 10% of the rated full thermal power pressure equivalent. This interlock is determined by one-out-of-two pressure detectors. The LCO requirement for this Function ensures that one of the inputs to the P-7 interlock is available.

The LCO requires two channels of Turbine Impulse Chamber Pressure, P-13 interlock to be OPERABLE in MODE 1 (1-out-of-2 coincidence).

The Turbine Impulse Chamber Pressure, P-13 interlock must be OPERABLE when the turbine generator is operating. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

19. Reactor Trip Breakers

This trip Function applies to the RTBs exclusive of individual trip mechanisms. The LCO requires two OPERABLE trains of trip breakers. A trip breaker train consists of the trip logic and all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the CRD Control Rod System. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single random failure can disable the RTS trip capability.

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19. Reactor Trip Breakers (continued)

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical (1-out-of-2 coincidence). In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs or associated bypass breakers are closed, and the GRD Control Rod System is capable of rod withdrawal or all rods are not fully inserted.

20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the GRD Rod Control System, or declared inoperable under Function 19 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical (1-out-of-2 coincidence). In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs and associated bypass breakers are closed, and the GRD Rod Control System is capable of rod withdrawal or all rods are not fully inserted.

21. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 19 and 20) and Automatic Trip Logic (Function 21) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is equipped with a bypass breaker to allow testing of the trip breaker while the unit is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE (1-out-of-2 coincidence). Having two OPERABLE channels ensures that random failure of a single logic channel will not prevent reactor trip.

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21. Automatic Trip Logic (continued)

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the RTBs and associated bypass breakers are closed, and the GRD Control Rod System is capable of rod withdrawal or all rods are not fully inserted.

22. Seismic Trip

The seismic trip system operates to shut down reactor operations should ground accelerations exceed a preset level in any two of the three orthogonal directions monitored (one vertical, two horizontal).

Three triaxial sensors (accelerometers) are anchored to the containment base in three separate locations 120 degrees apart. Each senses acceleration in three mutually orthogonal directions. Output signals are generated when ground accelerations exceed the preset level. These signals are transmitted to the Trains A and B Solid State Protection System (SSPS). If two of the three sensors in any direction produce simultaneous outputs, the logic produces trains A and B reactor trip signals.

The RTS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) the NRC Policy Statement.

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.1-1 are specified (e.g., on a per steam line, per loop, per basis), then the Condition may be entered separately for each steam line, loop, SG, etc. as appropriate.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

~~Reviewer's Note: Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use these times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.~~

(continued)

BASES

ACTIONS

A.1 (continued)

Condition A applies to all RTS protection Functions. Condition A addresses the situation where one or more required channels for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1, B.2.1, and B.2.2

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. This action addresses the train orientation of the SSPS for this Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the safety function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours (54 hours total time) followed by opening the RTBs within 1 additional hour (55 hours total time). The 6 additional hours to reach MODE 3 and the 1 hour to open the RTBs are reasonable, based on operating experience, to reach MODE 3 and open the RTBs exit the applicability from full power operation in an orderly manner and without challenging unit systems. With the RTBs open and the unit in MODE 3, Condition C is entered if the Manual Reactor Trip Function has not been restored and the Rod Control System is capable of rod withdrawal or all rods are not fully inserted this trip Function is no longer required to be OPERABLE.

C.1 and C.2.1, and C.2.2

Condition C applies to the following reactor trip Functions in MODE 3, 4, or 5 with the RTBs closed and the CRD Rod Control System capable of rod withdrawal or all rods not fully inserted:

(continued)

BASES

ACTIONS

C.1 and C.2.1 and C.2.2 (continued)

- Manual Reactor Trip;
- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

This action addresses the train orientation of the SSPS for these Functions. With one channel or train inoperable, the inoperable channel or train must be restored to OPERABLE status within 48 hours. If the affected Function(s) cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs must be opened rods must be fully inserted and the Rod Control System rendered incapable of rod withdrawal within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, rods fully inserted and the Rod Control System rendered incapable of rod withdrawal, these Functions are no longer required.

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

Condition C is modified by a Note stating that the transition from MODE 5 to MODE 5 with the Rod Control System capable of rod withdrawal or all rods not fully inserted is not permitted for Functions 19, 20, or 21. This Note specifies an exception to ECG 3.0.4 for this MODE 5 transition and avoids placing the plant in a condition where control rods can be withdrawn while the reactor trip system is degraded. This note is in addition to the requirements of ECG 3.0.4 which preclude the transition from either MODE 3 or MODE 4 to MODE 3 or MODE 4 with the Rod control System capable of rod withdrawal or all rods not fully inserted for Functions 19, 20, or 21 with one channel or train inoperable.

D.1.1, D.1.2, D.2.1, D.2.2, and D.3

Condition D applies to the Power Range Neutron Flux-High Function.

The NIS power range detectors provide input to the GRDRod Control System and the SG Water Level Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-10271-P-A (Ref. 7).

(continued)

BASES

ACTIONS

D.1.1, D.1.2, D.2.1, D.2.2, and D.3 (continued)

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to $\leq 75\%$ RTP within 12 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design limits. With one of the NIS power range detectors inoperable, 1/4 of the radial power distribution monitoring capability is lost.

As an alternative to the above actions, the inoperable channel can be placed in the tripped condition within 6 hours and the QPTR monitored once every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR every 12 hours compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued unit operation at power levels $\geq 75\%$ RTP. The 6 hour Completion Time and the 12 hour Frequency are consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Twelve hours are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 4 hour time limit is justified in Reference 7.

Required Action D.2.2 has been modified by a Note which ~~only requires states that SR 3.2.4.2 is not required to be performed until 12 hours after input from one of the Power Range Neutron Flux channel input to QPTR becomes inoperable and thermal power is $> 75\%$ RTP.~~ Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using ~~this the~~ movable incore detectors once per 12 hours may not be necessary.

(continued)

BASES

ACTIONS
(continued)

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux—Low;
- Overtemperature ΔT ;
- Overpower ΔT ;
- Power Range Neutron Flux—High Positive Rate;
- Power Range Neutron Flux—High Negative Rate;
- Pressurizer Pressure—High; and
- SG Water Level—Low Low; and
- ~~SG Water Level—Low coincident with Steam Flow/Feedwater Flow Mismatch.~~

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 4 hour time limit is justified in Reference 7.

(continued)

BASESACTIONS
(continued)F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, the overlap of the power range detectors, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

H.1

~~Condition H applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is below the P-6 setpoint and one or two channels are inoperable. Below the P-6 setpoint, the NIS source range performs the monitoring and protection functions. The inoperable NIS intermediate range channel(s) must be returned to OPERABLE status prior to increasing power above the P-6 setpoint. The NIS intermediate range channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10.~~

I.1

Condition I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

J.1

Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the GRD Rod Control System capable of rod withdrawal or all rods not fully inserted. With the unit in this Condition, below P-6, the

(continued)

BASES

ACTIONS

J.1 (continued)

NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition and the unit enters Condition L.

K.1 and K.2.1 and K.2.2

Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the RTBs closed and the CRD Rod Control System capable of rod withdrawal or all rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, 1 additional hour is allowed to fully insert all rods and place the Rod Control System in a condition incapable of rod withdrawal to open the RTB. Once these ACTIONS are completed the RTBs are open the core is in a more stable condition, and the unit enters Condition L. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour to open the RTBs, fully insert all rods and place the Rod Control System in a condition incapable of rod withdrawal, are justified in Reference 7.

L.1, L.2, and L.3

Condition L applies when the required number of OPERABLE Source Range Neutron Flux channels is not met in MODE 3, 4, or 5 with the RTBs open or with the Rod Control System incapable of rod withdrawal and all rods fully inserted. With the unit in this Condition, the NIS source range performs the monitoring and protection functions. With less than the required number of source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation. In addition to suspension of positive reactivity additions, all valves that could add unborated water to the RCS must be closed within 1 hour as specified in LCO 3.9.2. The isolation of unborated water sources will preclude a boron dilution accident.

Also, the SDM must be verified within 1 hour and once every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows

(continued)

BASES

ACTIONS

L.1, L.2, and L.3 (continued)

sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Required Action L.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Times of within 1 hour and once per 12 hours are based on operating experience in performing the Required Actions and the knowledge that unit conditions will change slowly.

M.1 and M.2

Condition M applies to the following reactor trip Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low;
- RCP Breaker Position (~~Two Loops~~);
- Undervoltage RCPs; and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint (~~above P-8 for the Reactor Coolant Flow-Low reactor trip function~~) ~~and below the P-8 setpoint~~. These Functions do not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below the P-7 setpoint. The 6 hours allowed to place the channel in the tripped condition is justified in Reference 7. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time. ~~The Reactor Coolant Flow - Low reactor trip function does not have to be OPERABLE below the P-8 setpoint; however, the Required Action must take the plant below the P-7 setpoint if an inoperable channel is not tripped within 6 hours, due to the shared components between this function and the Reactor Coolant Flow - Low trip function.~~

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant

(continued)

BASES
ACTIONS

M.1 and M.2 (continued)

OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition M.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

N.1 and N.2

~~NOT USED~~

~~Condition N applies to the Reactor Coolant Flow Low (Single Loop) reactor trip Function. With one channel inoperable, the inoperable channel must be placed in trip within 6 hours. If the channel cannot be restored to OPERABLE status or the channel placed in trip within the 6 hours, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours. This places the unit in a MODE where the LCO is no longer applicable. This trip Function does not have to be OPERABLE below the P-8 setpoint because other RTS trip Functions provide core protection below the P-8 setpoint. The 6 hours allowed to restore the channel to OPERABLE status or place in trip and the 4 additional hours allowed to reduce THERMAL POWER to below the P-8 setpoint are justified in Reference 7.~~

~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.~~

O.1 and O.2

~~NOT USED~~

~~Condition O applies to the RCP Breaker Position (Single Loop) reactor trip Function. There is one breaker position device per RCP breaker. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 6 hours. If the channel cannot be restored to OPERABLE status within the 6 hours, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours.~~

(continued)

BASES

ACTIONS

0.1 and 0.2 (continued)

~~This places the unit in a MODE where the LCO is no longer applicable. This Function does not have to be OPERABLE below the P-8 setpoint because other RTS Functions provide core protection below the P-8 setpoint. The 6 hours allowed to restore the channel to OPERABLE status and the 4 additional hours allowed to reduce THERMAL POWER to below the P-8 setpoint are justified in Reference 7. The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.~~

P.1 and P.2

Condition P applies to Turbine Trip on Low Fluid Auto Stop Oil Pressure or on Turbine Stop Valve Closure. With one or more channels inoperable, the inoperable channel must be placed in the trip condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring for the Turbine Trip on Turbine Stop Valve Closure function, where four-of-four channels are required to initiate a reactor trip, hence more than one channel may be placed in trip. For the Turbine Trip on Low Auto Stop Oil Pressure function, if one channel is placed in trip, only one additional channel is required to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours. The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 7.

The Required Actions have been modified by a Note that allows placing the an inoperable Low Auto Stop Oil Pressure channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

Q.1 and Q.2

Condition Q applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status (Required Action Q.1) or the unit must be placed in MODE 3 within the

(continued)

BASES

ACTIONS

Q.1 and Q.2 (continued)

next 6 hours. The Completion Time of 6 hours (Required Action Q.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours (Required Action Q.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows bypassing one train up to [4] hours for surveillance testing, provided the other train is OPERABLE.

R.1 and R.2

Condition R applies to the RTBs in MODES 1 and 2. These actions address the train orientation of the RTS for the RTBs. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

The Required Actions have been modified by ~~two~~ three Notes. Note 1 allows one ~~channel~~ RTB to be bypassed for up to 2 hours for surveillance testing or maintenance, provided the other ~~channel~~ train is OPERABLE. Note 2 allows one RTB to be bypassed ~~only for the time required for performing~~ for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms per Condition U if the other RTB train is OPERABLE. Note 3 allows one RTB to be bypassed for up to 4 hours for logic surveillance testing per Condition Q provided the other train is OPERABLE. The ~~2-hour~~ time limits are ~~is~~ justified in Reference 7 ~~5~~ and 13.

S.1 and S.2

Condition S applies to the P-6 and P-10 interlocks. With one ~~or more required channels inoperable~~ for one out of two or ~~two out of four coincidence logic~~, the associated interlock must be verified by observation of the associated ~~permissive annunciator window~~ to be in its required state for the existing unit condition

(continued)

BASES

ACTIONS

S.1 and S.2 (continued)

within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

T.1 and T.2

Condition T applies to the P-7, P-8, P-9, and P-13 interlocks. With ~~one or more required channel(s) inoperable for one out of two or two out of four coincidence logic~~, the associated interlock must be ~~verified by observation of the associated permissive annunciator window to be in its required state for the existing unit condition~~ within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

U.1, U.2.1, and U.2.2

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time) ~~followed by opening the RTBs in 1 additional hour (55 hours total time)~~. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

(continued)

BASES
ACTIONS

U.1, U.2.1, and U.2.2 (continued)

~~With the RTBs open and the unit in MODE 3, Condition C is entered if the inoperable trip mechanism has not been restored and the Rod Control System is capable of rod withdrawal or all rods are not fully inserted, this trip function is no longer required to be OPERABLE. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to restore the inoperable trip mechanism to OPERABLE status, consistent with Ref. 13, one of the diverse features. The allowable time for performing maintenance of the diverse features is 2 hours for the reasons stated under Condition R.~~

The Completion Time of 48 hours for Required Action U.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

V.1

NOT USED

~~With two RTS trains inoperable, no automatic capability is available to shut down the reactor, and immediate plant shutdown in accordance with LCO 3.0.3 is required.~~

W.1

~~Condition W applies to the Seismic Trip, in MODES 1 and 2. With one of the channels inoperable, START UP and/or POWER OPERATION may proceed provided the inoperable channel is placed in trip within the next 6 hours. If a direction is inoperable, then the channel must be considered inoperable. Placing the channel in the tripped condition creates a partial trip condition requiring only one out of two logic for actuation for that particular location.~~

~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 72 hours while performing routine surveillance testing of the other channels. The allowed 72 hour bypass time is reasonable based on the low probability of an event occurring while the channel is bypassed and on the time required to perform the required surveillance testing.~~

X.1

~~Condition X applies to the Trip Time Delay (TTD) circuitry for the SFG Water Level-Low Low trip function when THERMAL POWER is less than or equal to 50% RTP in MODES 1 and 2. With one or more TTD circuitry delay timers inoperable, adjust the threshold power level for no time delay to 0% RTP, or place the affected SFG-low low level in trip. The Completion Time of 6 hours is based on Reference 7.~~

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

~~Reviewer's Note: Certain Frequencies are based on approval topical reports. In order for a licensee to use these times, the licensee must justify the Frequencies as required by the staff SER for the topical report.~~

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.1

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output output power indications every 24 hours. If the calorimetric exceeds the NIS channel output power indications by > 2% RTP, the NIS is not declared inoperable, but the score channel gains must be adjusted consistent with the calorimetric power. If

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.2 (continued)

the NIS channel ~~output power indications~~ cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel ~~output power indications~~ shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel ~~output power indications~~ and the calorimetric is $> 2\%$ RTP. The second Note clarifies that this Surveillance is required only if reactor power is $\geq 15\%$ RTP and that ~~12~~ ~~24~~ hours is

~~allowed for performing the first Surveillance after reaching 15% RTP but prior to exceeding 30% RTP.~~ At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is $\geq 3\%$, the NIS channel is still OPERABLE, but must be readjusted.

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the $f(\Delta I)$ input to the overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is $\geq 3\%$. Note 2 clarifies that the Surveillance is required only if reactor power is ~~$\geq 50\%$~~ ~~[15%]~~ RTP and that 24 hours is allowed for performing the first Surveillance after reaching ~~50%~~ ~~[15%]~~ RTP.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, ~~since the slow changes in neutron flux are slow during the fuel cycle can be detected during this interval, the expected change in the absolute difference between the incore and excore AFD will be less than 3 percent AFD during this interval.~~

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.14. The bypass breaker test shall include a local manual shunt trip only. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition with the RTB bypass breaker installed, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function including operation of the P-7 permissive which is a logic function only. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the $f(\Delta I)$ input to the overtemperature ΔT Function.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.6 (continued)

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 50 75% RTP and that ~~[24] hours after achieving equilibrium conditions with thermal power >75% RTP is allowed for performing the first surveillance after reaching 50 75% RTP.~~

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every ~~[92]~~ days.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function.

Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

~~The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology trip setpoint value.~~

~~The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of Reference 7.~~

SR 3.3.1.7 is modified by two notes. a-Note 1 that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3. Note 2 requires that the quarterly COT for the source range instrumentation shall include verification by observation of the associated permissive annunciator window that the P-6 and P-10 interlocks are in their required state for the existing unit conditions.

The Frequency of ~~[92]~~ days is justified in Reference 7.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7 except and it is modified by a the same Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit conditions by observation of the associated permissive annunciator window. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 92 days of the Frequencies prior to reactor startup, 12 hours after reducing power below P-10 and four hours after reducing power below P-10 and P-6, as discussed below. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of "4 12 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup, 12 hours after reducing power below P-10 and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 for more than 12 hours or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 12 hour or 4 hour limit as applicable. Four hours is a reasonable time. These time limits are reasonable based on operating experience to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for the periods discussed above > 4 hours.

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every [92] days, as justified in Reference 7.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.9 (continued)

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every ~~[18]~~ months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific DCCP setpoint methodology. ~~The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.~~

The Frequency of ~~18~~ months is based on the ~~assumption of an 18-month assumed~~ calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every ~~[18]~~ months. This SR is modified by ~~a few three~~ Notes stating Note 1 states that neutron detectors are excluded from the CHANNEL CALIBRATION. Note 2 states that the test shall include verification that the time constants are adjusted to the prescribed values where applicable. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP. The CHANNEL CALIBRATION for the source range ~~and intermediate range~~ neutron detectors consists of obtaining the detector

(continued)

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SURVEILLANCE
REQUIREMENTSSR 3.3.1.11 (continued)

plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. For the intermediate range and power range channels, a test shall be performed that shows allowed variances of detector voltage do not effect detector operation. This Surveillance SR is also modified by Note 2.3 stating that this surveillance is not required to be performed until reactor power exceeds P-6 for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the [18] month Frequency. The source range plateau curves are obtained under the conditions that apply during a plant outage.

The 18 month Frequency is based on past operating experience, which has shown these components usually pass the Surveillance when performed on the 18 month Frequency. The conditions for obtaining the source range plateau curves and the power and intermediate range detector voltages are described above. The other remaining portions of the CHANNEL CALIBRATIONS may be performed either during a plant outage or during plant operation.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a CHANNEL CALIBRATION of the seismic trip, as described in SR 3.3.1.10, every [18] months. This SR is modified by a Note stating that this test shall include verification of the RCS resistance temperature detector (RTD) bypass loop flow rate.

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

This test will verify the rate lag compensation for flow from the core to the RTDs.

The Frequency is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

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SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.13

SR 3.3.1.13 is the performance of a COT of RTS interlocks every ~~18~~ months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, ~~Seismic Trip~~ and the SI Input from ESFAS. This TADOT is performed every ~~18~~ months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.15

SR 3.3.1.15 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to reactor startup. A Note states that this Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.

SR 3.3.1.16

SR 3.3.1.16 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in ~~Technical Requirements Manual, Section 15 (Ref. 8)~~ the FSAR (Ref. 1). Individual component response times are not modeled in the analyses.

(continued)

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SURVEILLANCE
REQUIREMENTS

SR 3.3.1.16 (continued)

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

The response time testing for the SG water level low-low does not include trip time delays. Response times include the transmitters, Eagle-21 process protection cabinets, solid state protection system cabinets, and actuation devices only. This reflects the response times necessary for THERMAL POWER in excess of 50 percent RTP. For those functions without a specified response time, SR 3.3.1.16 is not applicable.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic noise or power interrupt tests), (2) in-place, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632 P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" (Ref. 8) provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocations for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

As appropriate, each channel's response time must be verified every [18] months on a STAGGERED TEST BASIS. Each verification shall include at least one train such that both trains are verified at least once per 36 months. Testing of the final actuation devices is included in the testing verification. Response times cannot be

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SURVEILLANCE
REQUIREMENTS

SR 3.3.1.16 (continued)

determined during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.16 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input to the first electronic component in the channel.

SR 3.3.1.17

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST for the Seismic trip. The frequency of every 18 months is based on instrument reliability and operating history data.

REFERENCES

1. FSAR, Chapter 7.
2. FSAR, Chapter 6.
3. FSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. RTS/ESFAS Setpoint Methodology Study WCAP-11082, Rev. 2, Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Station - Egel 21 Version, May 1993
7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
8. Technical Requirements Manual, Section 15 FSAR, Chapter 7, "Response Times," WCAP 13632, PA-1, Rev. 2, "Elimination of Pressure Sensor Response Time Testing Requirements"
9. FSAR, Chapter 9.2.7 & 9.2.2
10. FSAR, Chapter 10.3 & 10.4
11. FSAR, Chapter 8.3
12. DCM S-38A, "Plant Protection System"
13. WCAP-13878, "Reliability of Potter & Brumfield MDR Relays" June 1994

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14. WCAP-13900, "Extension of Slave Relay Surveillance Test intervals", April 1994

15. WCAP-14117, "Reliability Assessment of Potter and Brumfield MDR Series Relays"

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured;
- Signal processing equipment including analog digital protection system, field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications; and
- Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable outputs from the signal process control and protection system.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). In some cases, the same channels also provide control system inputs. To account for calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the Trip Setpoint and Allowable

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BASES

BACKGROUND

Field Transmitters or Sensors (continued)

Values. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

Signal Processing Equipment

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in FSAR, Chapter {6} (Ref. 1), Chapter {7} (Ref. 2), and Chapter {15} (Ref. 3). If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, the main control board, the unit computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. In the case of the Digital Feedwater Control System (DFWCS), the median/signal select (MSS) feature prevents control/protection interaction even though there are only three inputs and 2-out-of-3 logic. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function

(continued)

BASES

BACKGROUND

Signal Processing Equipment (continued)

actuation. Again, a single failure will neither cause nor prevent the protection function actuation.

These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 2.

The channels are designed such that testing required to be performed at power may be accomplished without causing an ESF actuation.

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

The Trip Setpoints used in the bistables are based on the analytical limits stated in Reference 2. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.2-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the "RTS/ESFAS Setpoint Methodology Study" Study" WCAP-11082, Rev. 2, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Station - Eagle 21 Version" May 1993 (Ref. 6). The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

Each ~~Certain~~ channels can be tested on line to verify that the signal processing equipment and setpoint accuracy is within the specified allowance requirements of Reference 2. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SR section.

The Trip Setpoints and Allowable Values listed in Table 3.3.2-1 are based on the methodology described in Reference 6, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements.

The SSPS performs the decision logic for most ESF equipment actuation; generates the electrical output signals that initiate the required actuation; and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various

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BACKGROUND

Solid State Protection System (continued)

transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Each SSPS train has a built in testing device that can automatically test the decision logic matrix functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices. The master and slave relays are routinely tested to ensure operation. The test of the master relays energizes the relay, which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The SLAVE RELAY TEST actuates the devices if their operation will not interfere with continued unit operation. For the latter case, actual component operation is prevented by the SLAVE RELAY TEST circuit, and slave relay contact operation is verified by a continuity check of the circuit containing the slave relay. ~~For slave relays in the ESF actuation system circuit that are Potter & Brumfield type MDR relays, the SLAVE RELAY TEST is performed on a refueling frequency. The test frequency is based on relay reliability assessments presented in WCAP-13878 "Reliability Assessment of Potter and Brumfield MDR Series Relays," WCAP-13900 "Extension of Slave Relay Surveillance Test Intervals," and WCAP-14117, "Reliability Assessment of Potter and Brumfield MDR Series Relay." These reliability assessments are relay specific and apply only to Potter and Brumfield MDR series relays which are the only relays used in the ESF actuation system. Note that for normally energized applications, the relays may have to be replaced periodically in accordance with the guidance given in WCAP-13878 for MDR relays.~~

~~Reviewer's Note: No one unit ESFAS incorporates all of the Functions listed in Table 3.3.2-1. In some cases (e.g., Containment Pressure High 3, Function 2.c), the table reflects several different implementations of the same Function. Typically, only one of these implementations are used at any specific unit.~~

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure-Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3).

The LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped or bypassed during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to < 2200°F); and

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APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

1. Safety Injection (continued)
2. Boration to ensure recovery and maintenance of SDM ($k_{eff} < 1.0$).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Phase A Isolation;
- Containment Purge ~~Ventilation~~ Isolation;
- Reactor Trip;
- Turbine Trip ~~from Reactor Trip with P-9;~~
- Feedwater Isolation ~~and Feedwater Pump Turbine Trip;~~
- Start of motor driven auxiliary feedwater (AFW) pumps;
- ~~Control room ventilation isolation; to pressurization mode; and Auxiliary Building to Building and Safeguards or safeguards only mode;~~
- ~~Enabling automatic switchover of Emergency Core Cooling Systems (ECCS) suction to containment sump.~~
- ~~Start the diesel generators (DGs) and transfer to the startup bus;~~
- ~~Start the containment fan cooler units (CFCUs) in low speed;~~
- ~~Start the component cooling water and auxiliary salt water pumps;~~
- ~~Input to containment spray pump and discharge valve auto start (with containment spray signal);~~
- ~~Isolate SG sample blowdown lines~~

These other functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the turbine and reactor to limit power generation;
- Isolation of main feedwater (MFW) to limit secondary side mass losses;

(continued)

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- Start of AFW to ensure secondary side cooling capability;
- ~~Isolation of the control room to ensure habitability; and~~

(continued)

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1. Safety Injection (continued)

- ~~Enabling ECCS suction from the refueling water storage tank (RWST) switchover on low low RWST level to ensure continued cooling via use of the containment sump.~~
- ~~Start the DGs to compensate for a possible loss of offsite power (LOCALDOP)~~
- ~~Start the components associated with the accident heat removal systems~~

a. Safety Injection-Manual Initiation

The LCO requires one channel per train to be OPERABLE. The operator can initiate SI at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each channel consists of one ~~push-button control switch~~ and the interconnecting wiring to the actuation logic cabinet. Each ~~push-button control switch~~ actuates both trains. This configuration does not allow testing at power.

b. Safety Injection-Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Manual and automatic initiation of SI must be OPERABLE in MODES 1, 2, and 3. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Manual Initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but

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LCO, and
APPLICABILITY

b. Safety Injection-Automatic Actuation Logic and Actuation Relays
(continued)

because of the large number of components actuated on a SI, actuation is simplified by the use of the manual actuation push-buttons control switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation.

These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Unit pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

c. Safety Injection-Containment Pressure-High \pm

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure-High \pm provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters (d/p cells) and electronics are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment.

Thus, the high pressure Function will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

(continued)

BASES

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c. Safety Injection - Containment Pressure - High \pm
(continued)

Containment Pressure - High \pm must be OPERABLE in MODES 1, 2, 3 and 4 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

d. Safety Injection - Pressurizer Pressure - Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) relief or safety valve;
- SLB;
- A spectrum of rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer relief or safety valve;
- LOCAs; and
- SG Tube Rupture.

~~At some units Pressurizer pressure provides both control and protection functions: input to the Pressurizer Pressure Control System, reactor trip, and SI. Therefore, the actuation logic must be able to withstand both an input failure to control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic. For units that have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements.~~

(continued)

BASES

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d. Safety Injection - Pressurizer Pressure - Low
(continued)

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above P-11) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic SI actuation below this pressure setpoint is then performed by the Containment Pressure - High \pm signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

e. Safety Injection - Steam Line Pressure

(1) Steam Line Pressure - Low

Steam Line Pressure - Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure - Low provides an input to the DFACS any-control functions. The MSS function prevents the excursion of one of the inputs from causing a process disturbance that would require protective action from the remaining channels on the affected steam line. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective

(continued)

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(1) Steam Line Pressure - Low (continued)

requirements with a two-out-of-three logic on each steam line.

With the ~~some~~ transmitters typically located inside the steam tunnels penetration area, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

This Function is anticipatory in nature and has a typical lead/lag ratio of 50/5.

Steam Line Pressure - Low must be OPERABLE in MODES 1, 2, and 3 (above P-11) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern. Inside containment SLB will be terminated by automatic SI actuation via Containment Pressure - High \pm , and outside containment SLB will be terminated by the Steam Line Pressure - Negative Rate - High signal for steam line isolation. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to cause an accident.

(2) ~~NOT USED~~

~~Steam Line Pressure - High Differential Pressure Between Steam Lines~~

~~Steam Line Pressure - High Differential Pressure Between Steam Lines provides protection against the following accidents:~~

- ~~• SLB;~~
- ~~• Feed line break; and~~

(continued)

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(2) ~~Steam Line Pressure High Differential
Pressure Between Steam Lines~~ (continued)

- ~~Inadvertent opening of an SG relief or an SG safety valve.~~

~~Steam Line Pressure High Differential Pressure Between Steam Lines provides no input to any control functions. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the requirements, with a two out of three logic on each steam line.~~

~~With the transmitters typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties. Steam line high differential pressure must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). This function is not required to be OPERABLE in MODE 4, 5, or 6 because there is not sufficient energy in the secondary side of the unit to cause an accident.~~

f. g.

~~NOT USED~~

~~Safety Injection High Steam Flow in Two Steam Lines Coincident With T_{avg} Low Low or Coincident With Steam Line Pressure Low~~

~~These Functions (1.f and 1.g) provide protection against the following accidents:~~

- ~~SLB; and~~
- ~~the inadvertent opening of an SG relief or an SG safety valve.~~

~~Two steam line flow channels per steam line are required OPERABLE for these functions. The steam line flow channels are combined in a one out of~~

(continued)

BASES

APPLICABLE
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f. g. ~~Safety Injection High Steam Flow in Two Steam Lines Coincident With T_{avg} Low Low or Coincident With Steam Line Pressure Low~~ (continued)

~~two logic to indicate high steam flow in one steam line. The steam flow transmitters provide control inputs, but the control function cannot cause the events that the Function must protect against. Therefore, two channels are sufficient to satisfy redundancy requirements. The one out of two configuration allows online testing because trip of one high steam flow channel is not sufficient to cause initiation. High steam flow in two steam lines is acceptable in the case of a single steam line fault due to the fact that the remaining intact steam lines will pick up the full turbine load. The increased steam flow in the remaining intact lines will actuate the required second high steam flow trip. Additional protection is provided by Function 1.e.(2), High Differential Pressure Between Steam Lines.~~

~~One channel of T_{avg} per loop and one channel of low steam line pressure per steam line are required OPERABLE. For each parameter, the channels for all loops or steam lines are combined in a logic such that two channels tripped will cause a trip for the parameter. For example, for three loop units, the low steam line pressure channels are combined in two out of three logic. Thus, the Function trips on one out of two high flow in any two out of three steam lines if there is one out of one low low T_{avg} trip in any two out of three RCS loops, or if there is a one out of one low pressure trip in any two out of three steam lines. Since the accidents that this event protects against cause both low steam line pressure and low low T_{avg} , provision of one channel per loop or steam line ensures no single random failure can disable both of these Functions. The steam line pressure channels provide no control inputs. The T_{avg} channels provide control inputs, but the control function cannot initiate events that the Function acts to mitigate.~~

(continued)

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f. g. ~~Safety Injection High Steam Flow in Two Steam Lines Coincident With T_{avg} Low Low or Coincident With Steam Line Pressure Low~~

~~The Allowable Value for high steam flow is a linear function that varies with power level. The function is a ΔP corresponding to 44% of full steam flow between 0% and 20% load to 114% of full steam flow at 100% load. The nominal trip setpoint is similarly calculated.~~

~~With the transmitters typically located inside the containment (T_{avg}) or inside the steam tunnels (High Steam Flow), it is possible for them to experience adverse steady state environmental conditions during an SLB event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties. The Steam Line Pressure Low signal was discussed previously under Function 1.e.(1).~~

~~This Function must be OPERABLE in MODES 1, 2, and 3 (above P 12) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). This signal may be manually blocked by the operator when below the P 12 setpoint. Above P 12, this Function is automatically unblocked. This Function is not required OPERABLE below P 12 because the reactor is not critical, so feed line break is not a concern. SLB may be addressed by Containment Pressure High 1 (inside containment) or by High Steam Flow in Two Steam Lines coincident with Steam Line Pressure Low, for Steam Line Isolation, followed by High Differential Pressure Between Two Steam Lines, for SI. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to cause an accident.~~

(continued)

BASES

APPLICABLE
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(continued)

2. Containment Spray

Containment Spray coincident with an SI signal provides three primary functions:

1. Lowers containment pressure and temperature after an HELB in containment;
2. Reduces the amount of radioactive iodine in the containment atmosphere; and
3. Adjusts the pH of the water in the containment recirculation sump after a large break LOCA.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure;
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure; and
- Minimize corrosion of the components and systems inside containment following a LOCA.

The containment spray actuation signal coincident with SI starts the containment spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the containment spray pumps and mixed with a sodium hydroxide solution from the spray additive tank. When the RWST reaches the low level setpoint, the spray pumps suction are manually tripped and spray flow can then be shifted to the containment sump RHR system if continued containment spray is required. Containment spray is actuated manually by Containment Pressure High 3 or Containment Pressure - High High coincident with an SI signal.

a. Containment Spray - Manual Initiation

The operator can manually initiate containment spray at any time from the control room if an SI signal is present by simultaneously turning ~~two~~ both Containment Isolation Phase "B" / containment spray actuation Actuate Trains A & B switches in the same train. Because an inadvertent actuation of containment spray could have such serious consequences, two switches must be turned

(continued)

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a. Containment Spray—Manual Initiation (continued)

~~simultaneously and an SI signal must be present to initiate APPLICABILITY containment spray. There are two sets of two switches each in the control room. Simultaneously turning the two switches in either set will actuate containment spray in both trains in the same manner as the automatic actuation signal. Two Manual Initiation switches in each train are required to be OPERABLE. to ensure no single failure disables the Manual Initiation Function. Note that Manual Initiation of containment spray also actuates Phase B containment isolation and CVI.~~

b. Containment Spray—Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of containment spray must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for an accident to occur, and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. Manual initiation is also required in MODE 4, even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

(continued)

BASES

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APPLICABILITY
(continued)

c. Containment Spray - Containment Pressure

This signal provides protection against a LOCA or an SLB inside containment. The transmitters (d/p cells) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

This is one of the only Functions that requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

~~Two different logic configurations are typically used. Three and four loop units use four channels in a two-out-of-four logic configuration. This configuration may be called the Containment Pressure - High 3 Setpoint for three and four loop units, and Containment Pressure - High High Setpoint for other units. Some two loop units use three sets of two channels, each set combined in a one out of two configuration, with these outputs combined so that two out of three sets tripped initiates containment spray. This configuration is called Containment Pressure - High 3 Setpoint. Since containment pressure is not used for control, both of these arrangements exceed the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip. Containment Pressure - [High 3] [High High] must be OPERABLE in MODES 1, 2, 3 and 4 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary~~

(continued)

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c. Containment Spray - Containment Pressure
(continued)

sides to pressurize the containment and reach the
Containment Pressure - ~~High-3~~ (High High) setpoints.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except component cooling water (CCW), at a relatively low containment pressure indicative of primary or secondary system leaks. For these types of events, forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since CCW is required to support RCP operation, not isolating CCW on the low pressure Phase A signal enhances unit safety by allowing operators to use forced RCS circulation to cool the unit. Isolating CCW on the low pressure signal may force the use of feed and bleed cooling, which could prove more difficult to control.

Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation logic. All process lines penetrating containment, with the exception of CCW, are isolated. CCW is not isolated at this time to permit continued operation of the RCPs with cooling water flow to the thermal barrier heat exchangers and ~~air~~ oil coolers. All process lines not equipped with remote operated isolation valves are manually closed, or otherwise isolated, prior to reaching MODE 4.

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains. Note that manual

(continued)

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3. Containment Isolation (continued)

actuation of Phase A Containment Isolation also actuates Containment ~~Purge and Exhaust Ventilation~~ Isolation.

The Phase B signal isolates CCW. This occurs at a relatively high containment pressure that is indicative of a large break LOCA or an SLB. For these events, forced circulation using the RCPs is no longer desirable. Isolating the CCW at the higher pressure does not pose a challenge to the containment boundary because the CCW System is a closed loop inside containment. Although some system components do not meet all of the ASME Code requirements applied to the containment itself, the system is continuously pressurized to a pressure greater than the Phase B setpoint. Thus, routine operation demonstrates the integrity of the system pressure boundary for pressures exceeding the Phase B setpoint. Furthermore, because system pressure exceeds the Phase B setpoint, any system leakage prior to initiation of Phase B isolation would be into containment. Therefore, the combination of CCW System design and Phase B isolation ensures the CCW System is not a potential path for radioactive release from containment.

Phase B containment isolation is actuated by ~~Containment Pressure High 3~~ or Containment Pressure-High High, or manually, via the automatic actuation logic, as previously discussed. For containment pressure to reach a value high enough to actuate ~~Containment Pressure High 3~~ or Containment Pressure-High High, a large break LOCA or SLB must have occurred and containment spray must have been actuated. RCP operation will no longer be required and CCW to the RCPs is, therefore, no longer necessary. The RCPs can be operated with seal injection flow alone and without CCW flow to the thermal barrier heat exchanger.

Manual Phase B Containment Isolation is accomplished by the same switches that actuate Containment Spray. When the two switches ~~in either set~~ are ~~turned~~ operated simultaneously, Phase B Containment Isolation and Containment Spray will be actuated in both trains.

(continued)

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(continued)

a. Containment Isolation - Phase A Isolation

(1) Phase A Isolation - Manual Initiation

Manual Phase A Containment Isolation is actuated by either of two switches in the control room. Either switch actuates both trains. Note that manual initiation of Phase A Containment Isolation also actuates Containment Purge Ventilation Isolation.

(2) Phase A Isolation - Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of Phase A Containment Isolation must be OPERABLE in MODES 1, 2, 3 and 4, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a Phase A Containment Isolation, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase A Containment Isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase A Isolation - Safety Injection

Phase A Containment Isolation is also initiated by all Functions that initiate SI. The Phase A Containment Isolation

(continued)

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(3) Phase A Isolation-Safety Injection
(continued)

requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

b. Containment Isolation-Phase B Isolation

Phase B Containment Isolation is accomplished by Manual Initiation, Automatic Actuation Logic and Actuation Relays, and by Containment Pressure channels (the same channels that actuate Containment Spray, Function 2). The Containment Pressure trip of Phase B Containment Isolation is energized to trip in order to minimize the potential of spurious trips that may damage the RCPs.

(1) Phase B Isolation-Manual Initiation

(2) Phase B Isolation-Automatic Actuation Logic and Actuation Relays

Manual and automatic initiation of Phase B containment isolation must be OPERABLE in MODES 1, 2, and 3, and 4 when there is a potential for an accident to occur. ~~Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a Phase B containment isolation, actuation is simplified by the use of the manual actuation push buttons switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase B containment~~

(continued)

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- (1) Phase B Isolation - Manual Initiation
- (2) Phase B Isolation - Automatic Actuation Logic and Actuation Relays (continued)

isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

- (3) Phase B Isolation - Containment Pressure

The basis for containment pressure MODE applicability is as discussed for ESFAS Function 2.c above.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For an SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. ~~For units that do not have steam line check valves, Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.~~

a. Steam Line Isolation - Manual Initiation

Manual initiation of Steam Line Isolation can be accomplished from the control room ~~via an individual switch on each valve. There are two switches in the control room and either switch can initiate action to immediately close all MSIVs.~~ The LCO requires two ~~one~~ channels ~~per valve~~ to be OPERABLE.

(continued)

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(continued)

b. Steam Line Isolation - Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and ~~deactivated~~. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation - Containment Pressure - High 2 High

This Function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment to ~~maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment to a single SG.~~ The transmitters (d/p cells) are located outside containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure - High 2 ~~High~~ provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. However, for enhanced reliability, this Function was designed with four channels and a two-out-of-four logic. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions, and the Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure - High 2 ~~High~~ must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe

(continued)

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c. Steam Line Isolation—Containment Pressure—High 2
(continued)

break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed and ~~de-activated~~. ~~The actions~~ are taken to assure that the valve cannot be inadvertently opened. In MODE 4, the increase in containment pressure following a pipe break would occur over a relatively long time period such that manual action could reasonable be expected to provide protection and ESFAS Function 4.d need not be OPERABLE. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure—High 2 ~~High~~ setpoint.

d. Steam Line Isolation—Steam Line Pressure

(1) Steam Line Pressure—Low

Steam Line Pressure—Low provides closure of the MSIVs in the event of an SLB to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. This Function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven AFW pump. Steam Line Pressure—Low was discussed previously under SI Function 1.e.1.

Steam Line Pressure—Low Function must be OPERABLE in MODES 1, 2, and 3 (above P-11), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure—High 2 ~~High~~. Stuck valve transients and outside containment SLBs will be terminated by the Steam Line Pressure—Negative Rate—High signal for Steam Line Isolation below P-11 when SI has been manually blocked. The Steam Line Isolation Function is required in MODES 2

(continued)

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(1) Steam Line Pressure - Low (continued)

and 3 unless all MSIVs are closed and ~~deactivated~~. This function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

(2) Steam Line Pressure - Negative Rate - High

Steam Line Pressure - Negative Rate - High provides closure of the MSIVs for an SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure - Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure - Negative Rate - High signal is automatically enabled. Steam Line Pressure - Negative Rate - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy requirements with a two-out-of-three logic on each steam line.

Steam Line Pressure - Negative Rate - High must be OPERABLE in MODE 3 (2 per steam line) when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure - Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed and ~~deactivated~~. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

(continued)

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(2) Steam Line Pressure - Negative Rate - High
(continued)

While the transmitters may experience elevated ambient temperatures due to an SLB, the trip function is based on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

e. f. ~~NOT USED~~
~~Steam Line Isolation - High Steam Flow in Two Steam Lines Coincident with T_{sv9} - Low Low or Coincident With Steam Line Pressure - Low (Three and Four Loop Units)~~

~~These Functions (4.e and 4.f) provide closure of the MSIVs during an SLB or inadvertent opening of an SG relief or a safety valve, to maintain at least one unfaulted SG as a heat sink for the reactor and to limit the mass and energy release to containment.~~

~~These Functions were discussed previously as Functions 1.f. and 1.g.~~

~~These Functions must be OPERABLE in MODES 1 and 2, and in MODE 3, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines unless all MSIVs are closed and [de activated]. These Functions are not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.~~

g. ~~NOT USED~~
~~Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{sv9} - Low Low (Two Loop Units)~~

~~This Function provides closure of the MSIVs during an SLB or inadvertent opening of an SG relief or safety valve to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.~~

(continued)

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- g. ~~Steam Line Isolation High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} Low Low (Two Loop Units) (continued)~~

~~Two steam line flow channels per steam line are required OPERABLE for this Function. These are combined in a one out of two logic to indicate high steam flow in one steam line. The steam flow transmitters provide control inputs, but the control function cannot cause the events that the function must protect against. Therefore, two channels are sufficient to satisfy redundancy requirements. The one out of two configuration allows online testing because trip of one high steam flow channel is not sufficient to cause initiation.~~

~~The High Steam Flow Allowable Value is a ΔP corresponding to 25% of full steam flow at no load steam pressure. The Trip Setpoint is similarly calculated.~~

~~With the transmitters (d/p cells) typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoints reflect both steady state and adverse environmental instrument uncertainties.~~

~~The main steam line isolates only if the high steam flow signal occurs coincident with an SI and low low RCS average temperature. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2 1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.~~

~~Two channels of T_{avg} per loop are required to be OPERABLE. The T_{avg} channels are combined in a logic such that two channels tripped cause a trip for the parameter. The accidents that this Function protects against cause reduction of T_{avg} in the entire primary system. Therefore, the provision of two OPERABLE channels per loop in a~~

(continued)

BASES

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- g. ~~Steam Line Isolation - High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} - Low-Low (Two Loop Units) (continued)~~

~~two out of four configuration ensures no single random failure disables the T_{avg} - Low-Low Function. The T_{avg} channels provide control inputs, but the control function cannot initiate events that the function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues.~~

~~With the T_{avg} resistance temperature detectors (RTDs) located inside the containment, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrumental uncertainties.~~

~~This Function must be OPERABLE in MODES 1 and 2, and in MODE 3, when above the P-12 setpoint, when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. Below P-12 this Function is not required to be OPERABLE because the High High Steam Flow coincident with SI Function provides the required protection. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed and [de-activated]. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.~~

- h. ~~NOT USED
Steam Line Isolation - High High Steam Flow Coincident With Safety Injection (Two Loop Units)~~

~~This Function provides closure of the MSIVs during a steam line break (or inadvertent opening of a relief or safety valve) to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.~~

(continued)

BASES

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- h. ~~Steam Line Isolation High High Steam Flow
Coincident With Safety Injection (Two Loop Units)
(continued)~~

~~Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one out of two logic to indicate high steam flow in one steam line. The steam flow transmitters provide control inputs, but the control function cannot cause the events that the Function must protect against. Therefore, two channels are sufficient to satisfy redundancy requirements.~~

~~The Allowable Value for high steam flow is a ΔP , corresponding to 130% of full steam flow at full steam pressure. The Trip Setpoint is similarly calculated.~~

~~With the transmitters typically located inside the steam tunnels, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.~~

~~The main steam lines isolate only if the high steam flow signal occurs coincident with an SI signal. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.~~

~~This Function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines unless all MSIVs are closed and [de activated]. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.~~

(continued)

BASES

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(continued)

5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, and to stop the excessive flow of feedwater into the SGs. These functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

The Function is actuated when the level in any SG exceeds the high high setpoint, and performs the following functions:

- Trips the main turbine;
- Trips the MFW pumps;
- Initiates feedwater isolation; and
- Shuts the MFW regulating valves and the bypass feedwater regulating valves ~~coincident with P-4~~.

This Function is actuated by SG Water Level - High High. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated.

a. Turbine Trip and Feedwater Isolation - Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Turbine Trip and Feedwater Isolation - Steam Generator Water Level - High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level

(continued)

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- b. Turbine Trip and Feedwater Isolation - Steam Generator Water Level - High High (P-14)
(continued)

instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus, ~~four~~ three OPERABLE channels (narrow range instrument span each generator) are required to satisfy the requirements with a two-out-of-~~four~~ three logic and. ~~For units that have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. For other units that have only three channels, a median signal selector is provided to prevent control and protection function interactions or justification is provided in NUREG-1218 (Ref. 7).~~

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

- c. Turbine Trip and Feedwater Isolation - Safety Injection

Turbine Trip and Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

Turbine Trip and Feedwater Isolation Functions must be OPERABLE in MODES 1 and 2 ~~and 3~~ except when all MFIVs, MFRVs, ~~and associated bypass valves~~ are closed and ~~de-activated~~ or isolated by a closed manual valve when the MFW System is in operation and the turbine generator may be in operation. In MODES ~~3, 4, 5, and 6,~~ 3, 4, 5, and 6, the MFW System and the turbine

(continued)

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c. Turbine Trip and Feedwater Isolation - Safety Injection (continued)

generator are not in service and this Function is not required to be OPERABLE.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The system has two motor driven pumps and a turbine driven pump, making it available during normal unit operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break. The normal source of water for the AFW System is the condensate storage tank (CST) (normally not safety related). A low level in the CST will automatically realign the pump suction to the Essential Service Water (ESW) System (safety related). The AFW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately.

~~a. Auxiliary Feedwater - Manual Initiation~~

~~Manual initiation of Auxiliary Feedwater can be accomplished from the Control Room. Each of the three AFW pumps has a switch for manual initiation. The LCO requires three channels to be OPERABLE.~~

a ~~b~~. Auxiliary Feedwater - Automatic Actuation Logic and Actuation Relays (Solid State Protection System)

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

~~b ~~c~~. NOT USED~~

~~Auxiliary Feedwater - Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)~~

~~Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.~~

e ~~d~~. Auxiliary Feedwater - Steam Generator Water Level - Low Low

SG Water Level - Low Low provides protection against a loss of heat sink. A feed line break, inside or outside of containment, or a loss of MFW, would result in a loss of SG water level. SG Water Level - Low Low provides input to the SG

(continued)

BASES

(continued)

BASES

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e d. Auxiliary Feedwater - Steam Generator Water
Level - Low Low

Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system which may then require a protection function actuation and a single failure in the other channels providing the protection function actuation. Thus, ~~four~~ three OPERABLE channels (narrow range instrument span each generator) are required to satisfy the requirements with two-out-of-four ~~three~~ logic and ~~— For units that have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. For other units that have only three channels, a median signal selector is provided or justification is provided in Reference 7 for level control.~~

~~This function is actuated on two out of three low-low water level signals occurring in any steam generator. If a low-low water level condition is detected in one steam generator, signals are generated to start the motor driven auxiliary feedwater pumps. If a low-low water level condition is detected in two or more steam generators, a signal is generated to start the turbine driven auxiliary feedwater pump as well.~~

~~The signals to start auxiliary feedwater pumps are delayed through the use of a Trip Time Delay (TTD) system for reactor power levels below 50% of RTP. Low-low water level in any protection set in any steam generator will generate a signal which starts an elapsed time trip delay timer. The allowable trip time delay is based upon the prevailing power level at the time the low-low level trip setpoint is reached. If power level rises after the trip time delay setpoints have been determined, the trip time delay is re-determined (i.e., decreased) according to the increase in power level. However, the trip time delay is not changed if the power level decreases after the delay has been determined. The use of this delay allows added time for natural steam generator level stabilization or operator intervention to avoid an inadvertent protection system actuation.~~

With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

(continued)

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d ~~e~~. Auxiliary Feedwater - Safety Injection

An SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating functions and requirements.

e ~~f~~. ~~NOT USED~~ Auxiliary Feedwater - Loss of Offsite Power

~~A loss of offsite power to the service buses will be accompanied by a loss of reactor coolant pumping power and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each service 12 KV bus. Loss of power to either service bus will start the turbine driven AFW pumps to ensure that at least one SG contains enough water to serve as the heat sink for~~

(continued)

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e. Auxiliary Feedwater - Loss of Offsite Power
(continued)

~~reactor decay heat and sensible heat removal following the reactor trip.~~

Functions 6.a through ~~6.e~~ 6.b, 6.d, and 6.g must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor except the RCS AT time delays associated with Function 6.d, are only required to be operable in MODES 1 and 2. Below Mode 2, for the trip time delay, the maximum time delay is permitted; therefore, no OPERABILITY requirement is imposed on vessel AT channels in MODE 3. SG Water Level - Low Low in any operating SG will cause the motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level - Low Low in any two operating SGs will cause the turbine driven pumps to start. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

f. Auxiliary Feedwater - Undervoltage Reactor Coolant Pump

A loss of power on the buses that provide power to the RCPs provides indication of a pending loss of RCP forced flow in the RCS. The Undervoltage RCP Function senses the voltage ~~downstream~~ upstream of each RCP breaker. A loss of power, ~~or an open RCP breaker,~~ on two ~~or more~~ RCPs buses, will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

g. ~~h.~~ NOT USED

~~Auxiliary Feedwater - Trip of All Main Feedwater Pumps~~

~~A Trip of all MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal to bring the reactor back to no load temperature and pressure. A turbine~~

(continued)

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~~driven MFW pump is equipped with two pressure switches on the control air/oil~~

g ~~ii.~~ Auxiliary Feedwater Trip of All Main Feedwater Pumps (continued)

~~line for the speed control system. A low pressure signal from either of these pressure switches indicates a trip of that pump. Motor driven MFW pumps are equipped with a breaker position sensing device. An open supply breaker indicates that the pump is not running. Two OPERABLE channels per pump satisfy redundancy requirements with one out of two taken twice logic. A trip of all MFW pumps starts the motor driven and turbine driven AFW pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor.~~

~~Functions 6.f and 6.g must be OPERABLE in MODES 1 and 2. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 3, 4, and 5, the RCPs and MFW pumps may be normally shut down, and thus neither the pump trip is not indicative of a condition requiring automatic AFW initiation of the TDAFW pump. No other anticipatory start signals are necessary for the TDAFW pump, only low level in 2 of 4 SGs~~

h ~~ii.~~ NOT USED
Auxiliary Feedwater Pump Suction Transfer on Suction Pressure Low

~~A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal supply of water for the pumps, the CST. Two pressure switches are located on the AFW pump suction line from the CST. A low pressure signal sensed by any one of the switches will cause the emergency supply of water for both pumps to be aligned, or cause the AFW pumps to stop until the emergency source of water is aligned. ESW (safety grade) is then lined up to supply the AFW pumps to ensure an adequate supply of water for the AFW System to maintain at least one of the SGs as the heat sink for reactor decay heat and sensible heat removal.~~

~~Since the detectors are located in an area not affected by HELBs or high radiation, they will not experience any adverse environmental~~

(continued)

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~~h ii. Auxiliary Feedwater Pump Suction Transfer on
Suction Pressure Low (continued)~~

~~conditions and the Trip Setpoint reflects only steady
state instrument uncertainties.~~

~~This Function must be OPERABLE in MODES 1, 2, and 3 to
ensure a safety grade supply of water for the AFW
System to maintain the SGs as the heat sink for the
reactor. This Function does not have to be OPERABLE
in MODES 5 and 6 because there is not enough heat
being generated in the reactor to require the SGs as a
heat sink. In MODE 4, AFW automatic suction transfer
does not need to be OPERABLE because RHR will already
be in operation, or sufficient time is available to
place RHR in operation, to remove decay heat.~~

7. ~~Automatic Switchover to Containment Sump Residual Heat
Removal (RHR) Pump Trip - Refueling Water Storage Tank
(RWST) Low Level~~

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically manually switched to the containment recirculation sump. This pump trip feature is blocked if the RHR pumps are already taking suction from the Containment Recirculation sump. The low head residual heat removal (RHR) pumps and containment spray pumps draw the water from the containment recirculation sump, the RHR pumps pump the water through the RHR heat exchanger, inject the water back into the RCS, and supply the cooled water to the other ECCS pumps. Switchover from the RWST to the containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

(continued)

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(continued)

- a. ~~Automatic Switchover to Containment Sump -
Automatic Actuation Logic and Actuation Relays~~
- ~~Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.~~
- b, c. ~~Automatic Switchover to Containment Sump - Refueling Water Storage Tank (RWST) Level - Low Low Coincident With Safety Injection and Coincident With Containment Sump Level - High~~

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low low level in the RWST coincident with an SI signal provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with four ~~three~~ level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-~~three~~four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability.

The RWST - Low Low Allowable Value/Trip Setpoint has both upper and lower limits. The lower limit is selected to ensure switchover occurs before the RWST empties, to prevent ECCS pump damage. The upper limit is selected to ensure enough borated water is injected to ensure the reactor remains shut down. The high limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction.

The transmitters are located in an area not affected by HELBs or post accident high radiation. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

~~Automatic switchover occurs only if the RWST low low level signal is coincident with SI. This prevents accidental switchover during normal~~

(continued)

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~~b, c. Automatic Switchover to Containment
Sump Refueling Water Storage Tank (RWST)
Level Low Low Coincident With Safety Injection
and Coincident With Containment Sump Level High
(continued)~~

~~operation. Accidental switchover could damage ECCS pumps if they are attempting to take suction from an empty sump. The automatic switchover function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2 1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.~~

~~Reviewer's Note: In some units, additional protection from spurious switchover is provided by requiring a Containment Sump Level High signal as well as RWST Level Low Low and SI. This ensures sufficient water is available in containment to support the recirculation phase of the accident. A Containment Sump Level High signal must be present, in addition to the SI signal and the RWST Level Low Low signal, to transfer the suctions of the RHR pumps to the containment sump. The containment sump is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two out of four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability. The containment sump level Trip Setpoint/Allowable Value is selected to ensure enough borated water is injected to ensure the reactor remains shut down. The high limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction. The transmitters are located inside containment and thus possibly experience adverse environmental conditions. Therefore, the trip setpoint reflects the inclusion of both steady state and environmental instrument uncertainties.~~

~~Units only have one of the Functions, 7.b or 7.c.~~

(continued)

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~~b, c. Automatic Switchover to Containment
Sump Refueling Water Storage Tank (RWST)
Level Low Low Coincident With Safety Injection
and Coincident With Containment Sump Level High
(continued)~~

These ~~These~~ Functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. ~~These These~~ Functions are ~~is~~ not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

8. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

a. Engineered Safety Feature Actuation System Interlocks - Reactor Trip, P-4

The P-4 interlock is enabled when a reactor trip breaker (RTB) and its associated bypass breaker is open. ~~Once the P-4 interlock is enabled, automatic SI initiation is blocked after a [] second time delay.~~ This Function allows operators to ~~take manual control~~ manually block reactivation of SI systems after the initial phase of injection is complete. Once SI is blocked, automatic actuation of SI cannot occur until the RTBs have been manually closed. The functions of the P-4 interlock are:

(continued)

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a. Engineered Safety Feature Actuation System
Interlocks—Reactor Trip, P-4 (continued)

- Trip the main turbine;
- Isolate MFW with coincident low $T_{avg} \leq 554^{\circ}F$;
- ~~Prevent~~ ~~Allows manual block of the automatic~~ reactivation of SI after a manual reset of SI;
- Transfer the steam dump from the load rejection controller to the ~~unit~~ ~~plant~~ trip controller; and
- Prevent opening of the MFW ~~isolation~~ ~~reg~~ valves or ~~bypass~~ valves if they were closed on SI or ~~high~~ SG Water Level—High-High.

Each of the above Functions is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in generated ~~core~~ power. To avoid such a situation, the noted Functions have been interlocked with P-4 as part of the design of the unit control and protection system.

None of the noted Functions serves a mitigation function in the unit licensing basis safety analyses. Only the turbine trip Function is explicitly assumed since it is an immediate consequence of the reactor trip Function. Neither turbine trip, nor any of the other four Functions associated with the reactor trip signal, is required to show that the unit licensing basis safety analysis acceptance criteria are ~~not~~ ~~not~~ exceeded.

The RTB position switches that provide input to the P-4 interlock only function to energize or de-energize or open or close contacts. Therefore, this Function has no adjustable trip setpoint with which to associate a Trip Setpoint and Allowable Value.

(continued)

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a. Engineered Safety Feature Actuation System
Interlocks - Reactor Trip, P-4 (continued)

This Function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because the main turbine, the MFW System, and the Steam Dump System are not in operation.

b. Engineered Safety Feature Actuation System
Interlocks - Pressurizer Pressure, P-11

The P-11 interlock permits a normal unit cooldown and depressurization without actuation of SI or main steam line isolation. With two-out-of-three pressurizer pressure channels (discussed previously) less than the P-11 setpoint, the operator can manually block the Pressurizer Pressure-Low and Steam Line Pressure-Low SI signals and the Steam Line Pressure-Low steam line isolation signal (previously discussed). When the Steam Line Pressure-Low steam line isolation signal is manually blocked, a main steam isolation signal on Steam Line Pressure-Negative Rate-High is enabled. This provides protection for an SLB by closure of the MSIVs. With two-out-of-three pressurizer pressure channels above the P-11 setpoint, the Pressurizer Pressure-Low and Steam Line Pressure-Low SI signals and the Steam Line Pressure-Low steam line isolation signal are automatically enabled. The operator can also enable these trips by use of the respective manual reset buttons/switches. When the Steam Line Pressure-Low steam line isolation signal is enabled, the main steam isolation on Steam Line Pressure-Negative Rate-High is disabled. The Trip Setpoint reflects only steady state instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the actuation of SI or main steam isolation. This Function does not have to be OPERABLE in MODE 4.

(continued)

BASES (continued)

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- b. Engineered Safety Feature Actuation System Interlocks - Pressurizer Pressure, P-11
(continued)

5, or 6 because system pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

- c. ~~NOT USED~~ Engineered Safety Feature Actuation System Interlocks - T_{avg} - Low Low, P-12

~~On increasing reactor coolant temperature, the P-12 interlock reinstates SI on High Steam Flow Coincident With Steam Line Pressure Low or Coincident With T_{avg} - Low Low and provides an arming signal to the Steam Dump System. On decreasing reactor coolant temperature, the P-12 interlock allows the operator to manually block SI on High Steam Flow Coincident With Steam Line Pressure Low or Coincident with T_{avg} - Low Low. On a decreasing temperature, the P-12 interlock also removes the arming signal to the Steam Dump System to prevent an excessive cooldown of the RCS due to a malfunctioning Steam Dump System.~~

~~Since T_{avg} is used as an indication of bulk RCS temperature, this function meets redundancy requirements with one OPERABLE channel in each loop. In three loop units, these channels are used in two out of three logic. In four loop units, they are used in two out of four logic.~~

~~This function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This function does not have to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to have an accident.~~

The ESFAS instrumentation satisfies Criterion 3 of the NRC Policy Statement 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument Loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s)

(continued)

BASES

ACTIONS
(continued)

entered for the protection Function(s) affected. When the Required Channels in Table 3.3.2-1 are specified (e.g., on a per steam line, per loop, per SG, etc., basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

~~Reviewer's Note: Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use these times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.~~

A.1

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

(continued)

BASES

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(continued)

B.1, B.2.1 and B.2.2

Condition B applies to manual initiation of:

- SI;
- Containment Spray;
- Phase A Isolation; and
- Phase B Isolation.

This action addresses the train orientation of the SSPS for the functions listed above. If a channel or train is inoperable, 48 hours is allowed to return it to an OPERABLE status. Note that for containment spray and Phase B isolation, failure of one or both channels in one train renders the train inoperable. Condition B, therefore, encompasses both situations. The specified Completion Time is reasonable considering that there are two automatic actuation trains and another manual initiation train OPERABLE for each function, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (54 hours total time) and in MODE 5 within an additional 30 hours (84 hours total time). The allowable Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1 and C.2.2

Condition C applies to the automatic actuation logic and actuation relays for the following functions:

- SI;
- Containment Spray;
- Phase A Isolation;

(continued)

BASES

ACTIONS

C.1, C.2.1 and C.2.2 (continued)

- ~~Phase B Isolation; and~~
- ~~Automatic Switchover to Containment Sump.~~

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (12 hours total time) and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to [4] hours for surveillance testing, provided the other train is OPERABLE. This allowance is based on the reliability analysis assumption of WCAP-10271-P-A (Ref. 8) that 4 hours is the average time required to perform channel surveillance.

D.1, D.2.1, and D.2.2

Condition D applies to:

- ~~Containment Pressure High 1;~~
- Pressurizer Pressure - Low (~~two, three, and four loop units~~);
- Steam Line Pressure - Low;
- ~~Steam Line Differential Pressure High;~~
- ~~High Steam Flow in Two Steam Lines Coincident With T_{avg} - Low Low or Coincident With Steam Line Pressure - Low;~~

(continued)

BASES

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

- Containment Pressure - ~~High-2; High - High~~
- Steam Line Pressure - Negative Rate - High;
- ~~Steam Line Pressure - Low;~~
- ~~High Steam Flow Coincident With Safety Injection Coincident With T_{avg} - Low Low;~~
- ~~High High Steam Flow Coincident With Safety Injection;~~
- ~~High Steam Flow in Two Steam Lines Coincident With T_{avg} - Low Low;~~
- SG Water level - Low Low (two, three, and four loop units);
and

~~• SG Water level - High High (P-14) (two, three, and four loop units).~~

If one channel is inoperable, 6 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic (excluding ~~pressurizer pressure - low and containment pressure high - high~~). Therefore, failure of one channel places the Function in a two-out-of-two configuration. ~~One~~ ~~The inoperable~~ channel must be tripped to place the Function in a ~~one-out-of-three~~ ~~two~~ configuration that satisfies redundancy requirements.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel ~~or one additional channel~~ to be bypassed for up to ~~4~~ hours for surveillance testing of other channels. The 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the ~~4~~ hours allowed for testing, are justified in Reference 8.

(continued)

BASES

ACTIONS
(continued)

E.1, E.2.1, and E.2.2

Condition E applies to:

- ~~Containment Pressure - High~~
- Containment Spray Containment Pressure ~~High-3~~ (High, High) ~~(two, three, and four loop units); and~~
- Containment Phase B Isolation Containment Pressure - ~~High-3~~ (High, High),

None of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray. ~~The containment spray signal is also interlocked with SI and will not initiate without simultaneous SI and case spray signals.~~

To avoid the inadvertent actuation of containment spray and Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 6 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status, or place it in the bypassed condition within 6 hours, requires the unit be placed in MODE 3 within the following 6 hours ~~and MODE 4 within the next 6 hours and MODE 5 within 42 hours~~. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In ~~MODE 4~~, these Functions are no longer required OPERABLE.

ACTIONS

(continued)

BASES

E.1, E.2.1, and E.2.2 (continued)

The Required Actions are modified by a Note that allows one additional channel to be bypassed for up to 48 hours for surveillance testing. Placing a second channel in the bypass condition for up to 4 hours for testing purposes is acceptable based on the results of Reference 8.

F.1, F.2.1, and F.2.2

Condition F applies to:

- ~~Manual Initiation of Steam Line Isolation;~~
- ~~Loss of Offsite Power;~~
- ~~Auxiliary Feedwater Pump Suction Transfer on Suction Pressure Low; and~~
- P-4 Interlock.

~~For the Manual Initiation and the P-4 Interlock Functions, this action addresses the train orientation of the SSPS. For the Loss of Offsite Power Function, this action recognizes the lack of manual trip provision for a failed channel. For the AFW System pump suction transfer channels, this action recognizes that placing a failed channel in trip during operation is not necessarily a conservative action. Spurious trip of this function could align the AFW System to a source that is not immediately capable of supporting pump suction. If a train or channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of these Functions, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.~~

(continued)

BASES

ACTIONS
(continued)

G.1, G.2.1 and G.2.2

Condition G applies to the automatic actuation logic and actuation relays for the Steam Line Isolation [~~Turbine Trip and Feedwater Isolation.~~] and AFW actuation Functions.

The action addresses the train orientation of the SSPS and the master and slave relays for these functions. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to [4] hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 8) assumption that 4 hours is the average time required to perform channel surveillance.

H.1 and H.2

Condition H applies to the automatic actuation logic and actuation relays for the Turbine Trip and Feedwater Isolation Function.

This action addresses the train orientation of the SSPS and the master and slave relays for this function. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the following 6 hours. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of

(continued)

BASES

ACTIONS H.1 and H.2 (continued)

an event occurring during this interval. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. These Functions are no longer required in MODE 3. Placing the unit in MODE 3 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 8) assumption that 4 hours is the average time required to perform channel surveillance.

I.1 and I.2

Condition I applies to:

- SG Water Level -- High High (P-14) (two, three, and four loop units), and
- Undervoltage Reactor Coolant Pump.

If one channel of SG water level is inoperable, 6 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the Function is then in a partial trip condition where one-out-of-two or one-out-of-three logic will result in actuation. The 6 hour Completion Time is justified in Reference 8. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit to be placed in MODE 3 within the following 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, these Functions are no longer required OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, these Functions are no longer required OPERABLE.

If one channel of undervoltage reactor coolant pump is inoperable, 6 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the function is then in a

(continued)

BASES

ACTIONS

I.1 and I.2 (continued)

~~partial trip condition where one additional tripped channel will result in actuation. The 6 hour Completion Time is justified in Ref. 8. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the Unit to be placed in MODE 2 within the following 6 hours. The allowed Completion time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner without challenging unit systems. In MODE 2, this function is no longer required OPERABLE.~~

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to ~~4~~ hours for surveillance testing of other channels. The 6 hours allowed to place the inoperable channel in the tripped condition, and the 4 hours allowed for a second channel to be in the bypassed condition for testing, are justified in Reference 8.

J.1 and J.2

~~NOT USED~~

~~Condition J applies to the AFW pump start on trip of all MFW pumps.~~

~~This action addresses the train orientation of the SSPS for the auto start function of the AFW System on loss of all MFW pumps. The OPERABILITY of the AFW System must be assured by allowing automatic start of the AFW System pumps. If a channel is inoperable, 48 hours are allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours are allowed to place the unit in MODE 3. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above. The allowance of 48 hours to return the train to an OPERABLE status is justified in Reference 8.~~

K.1, K.2.1 and K.2.2

Condition K applies to:-

~~_____~~

- ~~• RWST Level Low Low Coincident with Safety Injection, and~~
- ~~• RWST Level Low Low Coincident with Safety Injection and Coincident with Containment Sump Level High.~~

(continued)

BASES

ACTIONS

K.1, K.2.1 and K.2.2 (continued)

~~RWST Level - Low, which trips both RHR pumps. Low Low Coincident With SI and Coincident With Containment Sump Level High provides actuation of switchover to the containment sump. Note that this Function requires the bistables to energize to perform their required action. The failure of up to two channels will not prevent the operation of this Function. However, placing a failed channel in the tripped condition could result in a premature switchover to the sump, prior to the injection of the minimum volume from the RWST. Placing the inoperable channel in bypass results in a two out of three logic configuration, which satisfies the requirement to allow another failure without disabling actuation of the switchover when required. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to ensure that the Function remains OPERABLE, and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high/low). Placing the out-of-service channel in bypass will generate a high level signal on that channel, which will ensure that under no circumstances can a failure of an additional channel low prevent the RHR pumps from starting as the result of an SI signal. The 6 hour Completion Time is justified in Reference 8. If the channel cannot be returned to OPERABLE status or placed in the bypass condition within 6 hours, and returned to an OPERABLE status within 72 hours, the unit must be brought to MODE 3 within the following 6 hours and MODE 5 within the next 30 hours immediately enter LCO 3.0.3. The 72 hour Allowed Outage Time (AOT) is the same AOT that is allowed for one inoperable RHR pump. This comparison is reasonable because the possible consequences of losing a second level channel can, in the worst case, be no more severe than the loss of one RHR pump, and the probability of losing the level channel is even lower than that of losing an RHR pump. The allowed Completion Times for shutdown are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the unit does not have any analyzed transients or conditions that require the explicit use of the protection/pump trip functions noted above.~~

The Required Actions are modified by a Note that allows placing a second channel in the bypass condition for up to 4 hours for surveillance testing. The total of 12 hours to reach MODE 3 and 4 hours for a second channel to be bypassed is acceptable based on the results of Reference 8.

(continued)

BASES

L.1, L.2.1 and L.2.2

Condition L applies to the P-11 and ~~P-12~~ [and ~~P-14~~] interlocks.

With one or more channels inoperable, the operator must verify that the interlock is in the required state for the existing unit condition. This action manually accomplishes the function of the interlock. Determination must be made within 1 hour. The verification determination can be made by observation of the associated annunciator window(s). The 1 hour Completion Time is equal to the time allowed by

(continued)

BASES

ACTIONS

L.1, L.2.1 and L.2.2 (continued)

LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of ESFAS function. If the interlock is not in the required state (or placed in the required state) for the existing unit condition, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of these interlocks.

M.1 or M.2

Condition M applies to the Trip Time Delay (TTD) for the SG low-low water level actuation of AFW pumps. With one or more TTD circuitry delay timers inoperable, 6 hours are allowed to adjust the threshold power level for no time delay to 0% RTP or to place the affected SG water level low-low channel in trip. The specified Completion Time is reasonable considering the nature of these functions, the available redundancy, and the low probability of an event occurring during this interval. If the TTD threshold power level cannot be adjusted or the affected SG water level low-low channel cannot be placed in trip, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above.

N.1 or N.2 and N.2.2

Condition N applies to

- Manual Initiation of Steam Line Isolation and
- Manual Initiation of Auxiliary Feedwater

If a channel is inoperable, 48 hours is allowed to return the channel to an OPERABLE status. The specified Completion Time is reasonable considering the nature of these functions, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the associated pump or valve shall be declared inoperable immediately and the REQUIRED ACTION of 3.7.5 or 3.7.2 as applicable complied with immediately.

(continued)

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The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

Note that each channel of process protection supplies both trains of the ESFAS. When testing channel I, train A and train B must be examined. Similarly, train A and train B must be examined when testing channel II, channel III, and channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

~~Reviewer's Note: Certain Frequencies are based on approved topical reports. In order for a licensee to use these times, the licensee must justify the Frequencies as required by the staff SER for the topical report.~~

SR 3.3.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read

(continued)

BASES

ACTIONS

SR 3.3.2.1 (continued)

approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff established in ~~STP I-1A~~, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.3

~~NOT USED~~

~~SR 3.3.2.3 is the performance of an ACTUATION LOGIC TEST as described in SR 3.3.2.2, except that the semiautomatic~~

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.3 (continued)

~~tester is not used and the continuity check does not have to be performed, as explained in the Note. This SR is applied to the balance of plant actuation logic and relays that do not have the SSPS test circuits installed to utilize the semiautomatic tester or perform the continuity check. This test is also performed every 31 days on a STAGGERED TEST BASIS. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.~~

SR 3.3.2.4

SR 3.3.2.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The time allowed for the testing (4 hours) and the surveillance interval are justified in Reference 8.

SR 3.3.2.5

SR 3.3.2.5 is the performance of a COT.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of the

(continued)

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SURVEILLANCE
REQUIREMENTS

SR 3.3.2.5 (continued)

surveillance interval extension analysis (Ref. 8) when applicable.

The Frequency of 92 days is justified in Reference 8.

SR 3.3.2.6

SR 3.3.2.6 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every ~~[92] days~~ 18 months. The Frequency is adequate, based on industry operating experience, considering instrument ~~relay~~ reliability and operating history data (Ref. 7)

SR 3.3.2.7 ~~NOT USED~~

~~SR 3.3.2.7 is the performance of a TADOT every [92] days. This test is a check of the Loss of Offsite Power, Undervoltage RCP, and AFW Pump Suction Transfer on Suction Pressure Low Functions. Each Function is tested up to, and including, the master transfer relay coils.~~

~~The test also includes trip devices that provide actuation signals directly to the SSPS. The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. The Frequency is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.~~

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.2.8

SR 3.3.2.8 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and ~~AFW pump start on trip of all MFW pumps.~~ It is performed every ~~[18]~~ months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.). The Frequency is adequate, based on industry operating experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a CHANNEL CALIBRATION.

A CHANNEL CALIBRATION is performed every ~~[18]~~ months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. ~~The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.~~

The Frequency of ~~[18]~~ months is based on the assumption of an ~~[18]~~ month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.2.10

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.3.2.10 (continued)

accident analysis. Response Time testing acceptance criteria are included in the ~~Technical Requirements Manual, Section 15 (Ref. 9) FSAR and SR 3.3.2.10 is only applicable to those functions with a specified limit.~~ Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

ESF RESPONSE TIME tests are conducted on an ~~[18]~~ month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every ~~[18]~~ months. The ~~[18]~~ month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching ~~[1000]~~ 650 psig in the SGs.

SR 3.3.2.11

SR 3.3.2.11 is the performance of a TADOT as described in SR 3.3.2.8, except that it is performed for the P-4 Reactor

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.11 (continued)

~~Trip Interlock, and the Frequency is once per RTB cycle. This
The 18 month Frequency is based on operating experience
demonstrating that undetected failure of the P-4 interlock
sometimes occurs when the RTB is cycled.~~

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

REFERENCES

1. FSAR, Chapter 6.
2. FSAR, Chapter 7.
3. FSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. ~~RTS/ESFAS Setpoint Methodology Study, WCAP-11082, Rev. 2,
Westinghouse Setpoint Methodology for Protection Systems
Diablo Canyon Stations - Eagle 21 Version, May 1993~~
7. ~~NUREG-1218, April 1988, WCAP-13900, "Extension of Slave
Relay Surveillance Test Intervals", April 1994~~
8. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
9. ~~Technical Requirements Manual, Section 15, "Response
Times - None - WCAP-13878, "reliability of Potter & Brumfield
MDR Relays", June 1994~~
10. ~~WCAP-14117, "Reliability Assessment of Potter and Brumfield
MDR Series Relays"~~

B 3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by unit specific documents in the FSAR section 7.5 (Ref. 1) addressing based upon the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category I variables.

Type A variables are included in this LCO because they provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for DBAs. ~~Because the list of Type A variables differs widely between units, Table 3.3.3-1 in the accompanying LCO contains no examples of Type A variables, except for those that may also be Category I variables.~~

Category I variables are the key variables deemed risk significant because they are needed to:

(continued)

BASES

BACKGROUND
(continued)

- Determine whether other systems important to safety are performing their intended functions;
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public, and to estimate the magnitude of any impending threat.

~~These key variables are identified by the unit specific Regulatory Guide 1.97 analyses (Ref. 1). These analyses identify the unit specific Type A and Category I variables and provide justification for deviating from the NRC proposed list of Category I variables.~~

~~Reviewer's Note: Table 3.3.3-1 provides a list of variables typical of those identified by the unit specific Regulatory Guide 1.97 analyses. Table 3.3.3-1 in unit specific Technical Specifications (TS) shall list all Type A and Category I variables identified by the unit specific Regulatory Guide 1.97 analyses, as amended by the NRC's Safety Evaluation Report (SER).~~

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

APPLICABLE
SAFETY ANALYSES

The PAM instrumentation ensures the operability of Regulatory Guide 1.97 Type A and Category I variables so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures (these variables are restricted to preplanned actions for the primary success path of DBAs), e.g., loss of coolant accident (LOCA);
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- Determine whether systems important to safety are performing their intended functions;
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of the NRC Policy Statement 10 CFR 50.36 (c)(2)(ii). Category I, non-Type A, instrumentation must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category I, non-Type A, variables are important for reducing public risk.

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category I, non-Type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident. This capability is consistent with the recommendations of Reference 1.

LCO 3.3.3 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident.

(continued)

BASES

LCO
(continued)

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. ~~More than two channels may be required at some units if the unit specific Regulatory Guide 1.97 analyses (Ref. 1) determined that failure of one accident monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function.~~

The exception to the two channel requirement is Containment Isolation Valve (CIV) Position, Auxiliary Feedwater (AFW) flow indication and Steam Generator (SG) water level (wide range). ~~In this case~~ For the CIV position, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

~~Both AFW flow indication and SG water level provide indication of the status of decay heat removal capability via the SGs. Sufficient water must be contained in the SGs in order to assure that heat removal can be accomplished through boiling in the SG. Although only one channel exists for each function, the channels provide diverse indication of the same capability.~~

~~Table 3.3.3-1 provides a list of variables typical of those identified by the unit specific Regulatory Guide 1.97 (Ref. 1) analyses. Table 3.3.3-1 in unit specific TS should list includes instrumentation which is classified as either all Type A and/or Category I variables in accordance with identified by the unit specific Regulatory Guide 1.97 analyses, as amended by the NRC's SER, FSAR Section 7.5, and SER 14.~~

Type A and Category I variables are required to meet Regulatory Guide 1.97 Category I (Ref. 2) design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display, ~~except as exempted in SSER 31. Regulatory Guide 1.97, for certain functions, requires that the function be recorded on at least one channel. For these channels where direct and immediate trend or transient information is not essential for operator information, or both channels would be recorded per Regulatory Guide 1.97, the loss of the recorder is not considered to be a loss of function. However, the recorder should be returned to service as soon as possible and an alternate means~~

(continued)

BASES

~~of obtaining the recorded information be established if the recorder is to be out-of-service beyond the channel AOT.~~

Listed below are discussions of the specified instrument Functions listed in Table 3.3.3-1. ~~These discussions are intended as examples of what should be provided for each Function when the unit specific list is prepared.~~

(continued)

BASES

LCO
(continued)

1. Power Range and Source Range Neutron Flux (Wide Range NIS)

~~Power Range and Source Range Neutron Flux~~ indication is provided to verify reactor shutdown. The ~~two ranges are wide range NIS~~ is necessary to cover the full range of flux that may occur post accident.

Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

2. Steam Line Pressure

~~Steam pressure is used to determine if a high energy secondary line rupture has occurred and the availability of the steam generators as a heat sink. It is also used to verify that a faulted steam generator is isolated. Steam pressure may be used to ensure proper cooldown rates or to provide a diverse indication for natural circulation cooldown.~~

3. 4. Reactor Coolant System (RCS) Hot and Cold Leg Temperatures

RCS Hot and Cold Leg Temperatures are Category I variables provided for verification of core cooling and long term surveillance.

RCS hot (~~outlet~~) and cold (~~inlet~~) leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin will allow termination of safety injection (SI), if still in progress, or reinitiation of SI if it has been stopped. RCS subcooling margin is also used for unit stabilization and cooldown control. RCS hot leg temperature also provides a temperature compensating signal for the reactor vessel level instrumentation system (RVLIS).

In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the unit conditions necessary to establish natural circulation in the RCS. The RCS cold leg temperature also provides a temperature input signal for the low temperature overpressure protection (LTOP) system.

Reactor outlet temperature inputs to the Reactor Protection System are provided by two fast response resistance elements and associated transmitters in each loop. The channels provide indication over a range of 32°F to 700°F.

(continued)

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5. Reactor Coolant System Pressure (Wide Range)

RCS wide range pressure is a Category I variable provided for verification of core cooling and RCS integrity long term surveillance.

RCS pressure is used to verify delivery of SI flow to RCS from at least one train when the RCS pressure is

(continued)

BASES

LCO

5. Reactor Coolant System Pressure (Wide Range) (continued)

below the pump shutoff head. RCS pressure is also used to verify closure of manually closed spray line valves and pressurizer power operated relief valves (PORVs).

In addition to these verifications, RCS pressure is used for determining RCS subcooling margin. RCS subcooling margin will allow termination of SI, if still in progress, or reinitiation of SI if it has been stopped. RCS pressure can also be used:

- to determine whether to terminate actuated SI or to reinitiate stopped SI;
- to determine when to reset SI and shut off low head SI;
- to manually restart low head SI;
- as reactor coolant pump (RCP) trip criteria; and
- to make a determination on the nature of the accident in progress and where to go next in the procedure.

RCS subcooling margin is also used for unit stabilization and cooldown control.

RCS pressure is also related to three decisions about depressurization. They are:

- to determine whether to proceed with primary system depressurization;
- to verify termination of depressurization; and
- to determine whether to close accumulator isolation valves during a controlled cooldown/depressurization.

A two final uses of RCS pressure is are to determine whether to operate the pressurizer heaters and as an input to Reactor Vessel Water Level Instrumentation System (RVLIS).

(continued)

BASES

LCO

5. Reactor Coolant System Pressure (Wide Range) (continued)

~~In some units, RCS pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture (SGTR) or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication. Furthermore, RCS pressure is one factor that may be used in decisions to terminate RCP operation.~~

6. Reactor Vessel Water Level

~~Reactor Vessel Water Level RVLS is provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy.~~

~~The Reactor Vessel Water Level Monitoring System RVLS provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory.~~

7. ~~a. Containment Wide Range Sump Water Level (Wide Range) and
b. Containment Recirculation Sump Level (Narrow Range)~~

~~Containment Wide Range Sump Water Level is provided for verification and long term surveillance of RCS integrity.~~

~~Containment Sump Water Level is used to determine:~~

- ~~• containment sump level accident diagnosis;~~
- ~~• when to begin the recirculation procedure; and~~
- ~~• whether to terminate SI, if still in progress.~~

~~Containment Recirculation Sump Level (Narrow Range) is used to verify that sufficient water is contained in the recirculation sump to allow operation of the Residual Heat Removal Pumps with the suction aligned to the containment recirculation sump. The required Regulatory Guide 1.97 recorder for this function is part of this instrument channel.~~

~~The Reactor Cavity Sump level instrumentation encompasses the range of the Containment Recirculation Sump and can be~~

(continued)

BASES

LCO
(continued)

~~Used to determine the appropriate time for swap-over of the RHR pumps from RWST to the Containment Recirculation Sump if required.~~

8. ~~a. Containment Pressure (Wide Range) and b. Containment Pressure (Normal Range)~~

~~Containment Pressure (Wide Range) is provided for verification of RCS and containment OPERABILITY.~~

~~Containment pressure is used to verify closure of main steam isolation valves (MSIVs) during a main steam line break inside containment, and containment spray Phase B isolation when High-3 high-high containment pressure is reached.~~

~~Both instruments are required to cover the Regulatory Guide 1.97 range requirements.~~

9. Containment Isolation Valve Position

~~CIV Position is provided for verification of Containment OPERABILITY, and Phase A and Phase B isolation, and containment ventilation system isolation.~~

~~When used to verify Phase A and Phase B isolation, the important information is the isolation status of the containment penetrations. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active CIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve, as applicable, and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE. This function is on a per valve basis and ACTION A is entered separately for each inoperable valve indication. Note (a) to the Required Channels states that the function is not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.~~

(continued)

BASES

LCO
(continued)

10. Containment Area Radiation (High Range)

Containment Area Radiation is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. Containment radiation level is used to determine if a high energy line break (HELB) containing radioactive fluid has occurred, and whether the event is inside or outside of containment.

11. Containment Hydrogen Concentration Monitors

Containment Hydrogen Monitors are Concentration monitoring is provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions and is used to determine whether or not hydrogen recombiners should be started.

12. Pressurizer Level

Pressurizer Level is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition.

13. a. Steam Generator Water Level (Wide Range) and b. Steam Generator Level (Narrow Range)

SG Water Level is provided to monitor operation of decay heat removal via the SGs. The Category I indication of SG level is the extended startup range level instrumentation. The extended startup wide range level covers a span of ≥ 6 inches to ≤ 394.582 inches above the lower tubesheet. The measured differential pressure is displayed in inches of water at 68°F.

Temperature compensation of this indication is performed manually by the operator. Redundant monitoring capability is provided by two trains of instrumentation. The uncompensated level signal is

(continued)

BASES

LCO

13. Steam Generator Water Level (Wide Range) (continued)

~~input to the unit computer, a control room indicator, and the Emergency Feedwater Control System.~~

SG Water Level ~~(Wide Range)~~ is used to:

- identify the faulted SG following a tube rupture;
- verify that the intact SGs are an adequate heat sink for the reactor;
- determine the nature of the accident in progress (e.g., verify an SGTR); and
- verify unit conditions for termination of SI during secondary unit HELBs outside containment.

~~At some units, Operator action is based on the control room indication of SG level. The RCS response during a design basis small break LOCA depends on the break size. For a certain range of break sizes, the boiler condenser reflux cooling mode of heat transfer is necessary to remove decay heat. Extended startup wide range level is a Type A variable because the operator must manually raise and control SG level to establish boiler condenser reflux cooling heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated extended startup wide range level reaches the boiler condenser setpoint reflux cooling initiation point.~~

~~SG Water Level (Narrow Range) is redundant to the SG wide range level and provides indication of adequate RCS heat removal capability during normal SG inventory conditions. The narrow range level covers a span from 437 inches to 581 inches above the lower tubesheet.~~

14. Condensate Storage Tank (CST) Level

CST Level is provided to ensure water supply for auxiliary feedwater (AFW). The CST provides the ensured safety grade water supply for the AFW System. ~~The CST consists of two identical tanks connected by a common outlet header. Inventory is monitored by a 0 inch to 144 inch level indication for each tank. CST Level is displayed on a control room indicator, strip chart recorder, and unit computer. In addition, a control room annunciator alarms on low level.~~

(continued)

BASES

LCO

14. Condensate Storage Tank (CST) Level (continued)

~~At some units, CST Level is considered a Type A variable because the control room meter and annunciator are is considered the primary indication used by the operator.~~

The DBAs that require AFW are the loss of electric power, steam line break (SLB), and small break LOCA.

The CST is the initial source of water for the AFW System. However, as the CST is depleted, manual operator action is necessary to replenish the CST or align suction to the AFW pumps from the hotwell ~~Fire Water Storage Tank or other alternate sources.~~

15, 16, 17, 18. In-Core Exit Temperature Thermocouples

~~In-Core Exit Temperature is Thermocouples are provided for verification and long term surveillance of core cooling.~~

An evaluation was made of the minimum number of valid core exit ~~in-core~~ thermocouples (GET) necessary for measuring core cooling. The evaluation determined the reduced complement of GETs ~~in-core thermocouple~~ necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities, including incore effects of the radial decay power distribution, excore effects of condensate runback in the hot legs, and nonuniform inlet temperatures. Based on these evaluations, ~~adequate core cooling is ensured can be adequately monitored with two valid Core Exit Temperature in-core thermocouple channels per quadrant with two GETs in-core thermocouples per required channel. The GET pair are oriented radially to permit evaluation of core radial decay power distribution. Core Exit Temperature is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.~~

~~Two OPERABLE channels of Core Exit Temperature In-Core Thermocouples are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Therefore, two randomly selected thermocouples are not sufficient to~~

(continued)

BASES

LCO 15, 16, 17, 18. Core Exit Temperature (continued)

~~meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one is located near the center of the core and the other near the core perimeter, such that the pair of Core Exit Temperatures indicate the radial temperature gradient across their core quadrant. Unit specific evaluations in response to Item II.F.2 of NUREG 0737 (Ref. 3) should have identified the thermocouple pairings that satisfy these requirements. Two sets of two thermocouples ensure a single failure will not disable the ability to determine the radial temperature gradient.~~

19. Auxiliary Feedwater (AFW) Flow

AFW Flow is provided to monitor operation of decay heat removal via the SGs.

The AFW Flow to each SG is determined from a differential pressure measurement calibrated for a range of 0 gpm to 1200 ~~300~~ gpm. ~~Redundant monitoring capability is provided by two independent trains of instrumentation for each SG.~~ Each differential pressure transmitter provides an input to a control room indicator and the unit computer. Since the primary indication used by the operator during an accident is the control room indicator, the PAM specification deals specifically with this portion of the instrument channel.

AFW flow is used three ways:

- to verify delivery of AFW flow to the SGs;
- to determine whether to terminate SI if still in progress, in conjunction with SG water level (narrow range); and
- to regulate AFW flow so that the SG tubes remain covered.

(continued)

BASES

LCO

19. Auxiliary Feedwater Flow (continued)

~~At some units, AFW flow is a Type A variable because operator action is required to throttle flow during an SLB accident to prevent the AFW pumps from operating in runout conditions. AFW flow is also used by the operator to verify that the AFW System is delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level (Narrow Range) during normal SG inventory conditions.~~

~~20. (new) Refueling Water Storage Tank (RWST) Water Level~~

~~RWST water level is used to verify the water source availability to the emergency core cooling system (ECCS) and Containment Spray Systems. It may also provide an indication of time for initiating cold leg recirculation from the sump following a LOCA. The RWST level signal trips the Residual Heat Removal Pumps at 33% in preparation for transfer to cold leg recirculation.~~

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3 ~~except for the Containment hydrogen Concentration monitor that is only required to be OPERABLE in MODES 1 and 2.~~ These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. ~~In MODES 4, 5, and 6, and in MODE 2 for the Containment Hydrogen Concentration monitor, unit conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.~~

ACTIONS

Note 1 has been added in the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident using alternate instruments and methods, and the low probability of an event requiring these instruments.

Note 2 has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1.

(continued)

BASES

ACTIONS

A.1 (continued)

The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

Condition A applies when one or more Functions have one required channel that is inoperable, ~~but at least one OPERABLE remaining channel~~. Required Action A.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

Condition B applies when the Required Action and associated Completion Time for Condition A are not met. This Required Action specifies initiation of actions in Specification 5.6.8, which requires a written report to be submitted to the NRC immediately. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of unit conditions that would require information provided by this instrumentation.

C.1

Condition C applies when one or more Functions have ~~two inoperable required no OPERABLE~~ channels (i.e., ~~two channels inoperable in the same Function~~). Required Action C.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with ~~two no~~ required channels ~~inoperable~~ OPERABLE in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration

(continued)

BASES

ACTIONS

C.1 (continued)

of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur. Condition C is modified by a Note that excludes hydrogen monitor channels.

D.1

Condition D applies when two hydrogen monitor channels are inoperable. Required Action D.1 requires restoring one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable based on the backup capability of the Post Accident Sampling System to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA (which would cause core damage) would occur during this time. Condition D is modified by a Note that limits the APPLICABILITY for the Containment Hydrogen Concentration monitor to MODES 1 and 2.

E.1

Condition E applies when the Required Action and associated Completion Time of Condition C or D are not met. Required Action E.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C or D, and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1 and F.2

If the Required Action and associated Completion Time of Conditions C or D are not met and Table 3.3.3-1 directs entry into Condition F, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

from full power conditions in an orderly manner and without challenging unit systems.

G.1

~~At this unit, Alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation have been will be developed and tested demonstrated prior to use. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the unit but rather to follow the directions of Specification 5.6.8, in the Administrative Controls section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.~~

H.1

~~If the Required Action and associated Completion Time of Conditions D is not met and Table 3.3.3-1 directs entry into Condition H, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.~~

~~The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.~~

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus,

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.3.1 (continued)

it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized. The Containment Hydrogen Concentration monitors are maintained in a "standby" condition which does not energize all of the monitor components, thus the monitors are not considered "normally energized".

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every [18] months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter with the necessary range and accuracy. This SR is modified by a two Notes that Note 1 excludes neutron detectors from CHANNEL CALIBRATION. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." Note 2 discusses an allowed methodology for calibrating the Containment Radiation Level (High Range) Function. The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. [Unit specific document (e.g., FSAR, NRC Regulatory Guide 1.97 SER letter).] 7.5
2. Regulatory Guide 1.97, [date] Revision 3.
3. NUREG-0737, Supplement 1, "TMI Action Items."

(continued)

BASES

4. Supplemental Safety Evaluation Report 14

5. Supplemental Safety Evaluation Report 31

B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator (SG) safety valves or the SG atmospheric dump valves (ADVs) can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System and the ability to borate the Reactor Coolant System (RCS) allows extended operation in MODE 3 until such time that either control is transferred back to the Control Room or a cooldown is initiated from outside the control room.

If the control room becomes inaccessible, the operators can establish control at the remote shutdown panel (hot shutdown panel), and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located at the remote hot shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the remote shutdown control and instrumentation functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown System is required to Instrumentation Functions and the hot shutdown panel controls provides equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 3.

The criteria governing the design and specific system requirements of the Remote Shutdown System Instrumentation Functions and controls are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The Remote Shutdown System Instrumentation Functions and the hot shutdown panel controls is considered an important contributor to the reduction of unit risk to accidents and as such it has been retained in the Technical Specifications as indicated in the NRC Policy Statement by Criterion 4 of 10 CFR 50.36(c)(2)(iii).

LCO

The Remote Shutdown System Instrumentation Functions and the hot shutdown panel controls LCO provides the OPERABILITY requirements of the instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation and controls typically required are listed in Table 3.3.4-1 in the accompanying LCO.

~~Reviewer's Note: For channels that fulfill GDC 19 requirements, the number of OPERABLE channels required depends upon the unit licensing basis as described in the NRC unit specific Safety Evaluation Report (SER). Generally, two divisions are required OPERABLE. However, only one channel per a given Function is required if the unit has justified such a design, and NRC's SER accepted the justification.~~

The controls, instrumentation, and transfer switches are required for:

- ~~Core reactivity control (initial and long term) Reactor trip indication;~~
- RCS pressure control;
- Decay heat removal via the AFW System and the SG safety valves or SG ADVs;
- RCS inventory control via charging flow; and
- Safety support systems for the above Functions, including service water auxiliary saltwater, component cooling water, and onsite power, including the diesel generators.

A Function of a Remote Shutdown System is OPERABLE if all required instrument and control channels needed to support the Remote Shutdown System Function for that function listed in Table 3.3.4-1 are OPERABLE. In some cases, Table 3.3.4-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Function is OPERABLE as long

(continued)

BASES

LCO
(continued) as one channel of any of the alternate information or control sources is OPERABLE.

The remote shutdown instrument and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments and control circuits will be OPERABLE if unit conditions require that the Remote Shutdown System be placed in operation.

APPLICABILITY The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room until either control is transferred back to the control room or a cooldown is initiated.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the facility is already subcritical and in a condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument control functions if control room instruments or controls become unavailable.

ACTIONS

Note 1 is included which excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the low probability of an event requiring the Remote Shutdown System and because the equipment can generally be repaired during operation without significant risk of spurious trip.

Note 2 has been added to the ACTIONS to clarify the application of Completion Time rules. Separate Condition entry is allowed for each Function listed on Table 3.3.4-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

ACTIONS
(continued)

A.1

Condition A addresses the situation where one or more required Functions of the Remote Shutdown System Instrumentation and SD panel controls are inoperable. This includes any Function listed in Table 3.3.4-1, as well as the control and transfer switches.

The Required Action is to restore the required Function to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.3.4.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

A channel check of the RTBs is inappropriate since their indication is local and any gross failure would be readily detected.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If the channels are

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.4.1 (continued)

~~within the criteria, it is an indication that the channels are OPERABLE. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are normally off scale during times when Surveillance is required, the CHANNEL CHECK will verify only that they are off scale in the same direction. Offscale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.~~

~~As specified in the Surveillance, a CHANNEL CHECK is only required for those channels which are normally energized.~~

The Frequency of 31 days is based upon operating experience which demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.4.2

SR 3.3.4.2 verifies each required Remote Shutdown System control circuit and transfer switch performs the intended function. This verification is performed from the ~~remote~~ ^{hot} shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. The ~~{18}~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. (However, this Surveillance is not required to be performed only during a unit outage.) Operating experience demonstrates that remote shutdown control channels usually pass the Surveillance test when performed at the ~~{18}~~ month Frequency.

SR 3.3.4.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

~~The channel calibration is not applicable to the RTB indication~~

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.4.3 (continued)

The Frequency of [18] months is based upon operating experience and consistency with the typical industry refueling cycle.

SR 3.3.4.4

~~SR 3.3.4.4 is the performance of a TADOT every 18 months. This test should verify the OPERABILITY of the reactor trip breakers (RTBs) open and closed indication on the remote shutdown panel, by actuating the RTBs. The Frequency is based upon operating experience and consistency with the typical industry refueling outage.~~

~~NOTE: A surveillance of the reactor trip breaker OPERABILITY is not required as part of the SURVEILLANCE REQUIREMENT since a TRIP ACTUATING DEVICE OPERATIONAL TEST of the reactor trip breakers is performed as part of the SURVEILLANCE REQUIREMENT for TS 3.3.1.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.
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B 3.3 INSTRUMENTATION

B 3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to degraded below a point that would allow safe unit operation. Undervoltage protection will generate an LOP start if a loss of voltage or degraded voltage condition occurs in on the switchyard 4.16KV vital bus. There are two three LOP start signals, one for each 4.16 kV vital bus.

Three undervoltage relays with inverse time characteristics are provided on each 4160 Class 1E instrument vital bus for detecting a sustained degraded voltage condition or a loss of bus voltage. The relays are combined in a two-out-of-three logic to will generate an LOP signal (first level undervoltage type relay setpoint) if the voltage is below 75% for a short time or below 90% for a long time. The DG start relays (one per bus) have an inverse time characteristic and will generate an LOP signal with a ≤ 0.8 sec time delay at ≥ 0 volts and at ≤ 10 seconds for ≥ 2583 volts. In addition, the circuit breakers for all loads, except the 4160-480 V load center transformers, are opened automatically by a similar set of first level undervoltage relays. Each of the vital 4160 kV buses has a separate pair of these relays. The relays have a two-out-of-two logic arrangement for each bus to prevent inadvertent tripping of operating loads during a loss of voltage either from a single failure in the potential circuits or from human error. One relay trips instantaneously at ≥ 2870 volts. The second of the two relays has an inverse time characteristic and a delay of ≤ 4 seconds at no voltage and a ≤ 25 second delay with ≥ 2583 volts to prevent loss of operating loads during transient voltage dips, and to permit the offsite power sources to pick up the load. The LOP start actuation is described in FSAR, Section 8.3 (Ref. 1).

Should there be a degraded voltage condition where the voltage of the vital 4160 kV buses remains at approximately 3785 kV or below, but above the setpoints of the first level undervoltage relays, the following second level undervoltage actions occur automatically:

- (1) After a ≤ 10 second time delay, the respective diesel generators will start.
- (2) After a ≤ 20 second time delay, if the undervoltage condition persists, the circuit breakers for all loads to the respective vital 4160 kV buses, except the 4160-480 V load center transformer, are opened and sequentially loaded on the DG.

Each vital 4160 kV bus has two second level undervoltage relays and one associated timer to initiate each of the above actions (1) and (2) (one timer for each action).

(continued)

BASES

Trip Setpoints and Allowable Values

The Trip Setpoints used in the relays are based on the analytical limits presented in FSAR, Chapter 15 (Ref. 2). The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account.

The actual nominal Trip Setpoint entered into the relays is normally still more conservative than that required by the Allowable Value. If the measured setpoint does not exceed the Allowable Value, the relay is considered OPERABLE.

Setpoints adjusted in accordance with the Allowable Value ensure that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the accident and that the equipment functions as designed.

Allowable Values and/or Trip Setpoints are specified for each Function in the LCO. Nominal Trip Setpoints are also specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by the surveillance procedure does not exceed the Allowable Value if the relay is performing as required. If the measured setpoint does not exceed the Allowable Value, the relay is considered OPERABLE. Operation with a Trip Setpoint less conservative than the nominal Trip Setpoint, but within the Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculation. Each Allowable Value and/or Trip Setpoint specified is more conservative than the analytical limit assumed in the transient and accident analyses in order to account for instrument uncertainties appropriate to the trip function. These uncertainties are defined in the "Unit Specific RTS/ESFAS Setpoint Methodology Study" WCAP 11082, Rev. 2 Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Stations - Eagle 21 Version (Ref. 3).

APPLICABLE SAFETY ANALYSES The LOP DG start instrumentation is required for the Engineered Safety Features (ESF) Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS).

Accident analyses credit the loading of the DG based on the loss of offsite power during a loss of coolant accident (LOCA). The actual DG start has historically been associated with the ESFAS actuation. The DG loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. The analyses assume a non-mechanistic DG loading, which does not explicitly account for each individual component of loss of power detection and subsequent actions.

(continued)

BASES

The required channels of LOP DG start instrumentation, in conjunction with the ESF systems powered from the DGs, provide unit protection in the event of any of the analyzed accidents discussed in Reference 2, in which a loss of offsite power is assumed.

The delay times assumed in the safety analysis for the ESF equipment include the 10 second DG start delay, and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," include the appropriate DG loading and sequencing delay.

(continued)

BASES

APPLICABLE SAFETY ANALYSES The LOP DG start instrumentation channels satisfy Criterion 3 of the NRC Policy Statement.
(continued)

LCO The LCO for LOP DG start instrumentation requires that ~~three~~ one channels per bus of both the loss of voltage and two channels per bus for initiation of load shed and two channels per bus of degraded voltage with one timer per bus for DG start and initiation of load shed Functions shall be OPERABLE in MODES 1, 2, 3, and 4 when the LOP DG start instrumentation supports safety systems associated with the ESFAS. In MODES 5 and 6, the ~~three~~ channels must be OPERABLE whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed. Loss of the LOP DG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power the DG powers the motor driven auxiliary feedwater pumps. Failure of these pumps to start would leave only one turbine driven pump, as well as an increased potential for a loss of decay heat removal through the secondary system.

APPLICABILITY The LOP DG Start Instrumentation Functions are required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE so that it can perform its function on an LOP or degraded power to the vital bus.

ACTIONS In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the channel is found inoperable, then the function that channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected.

Because the required channels are specified on a per bus basis, the Condition may be entered separately for each bus as appropriate.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of

(continued)

BASES

ACTIONS
(continued)

this Specification may be entered independently for each Function listed in the LCO. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

~~Condition A applies to the LOP DG start Function with one loss of voltage or degraded voltage channel per bus inoperable.~~

~~If one channel is inoperable, Required Action A.1 requires that channel to be placed in trip within 6 hours. With a channel in trip, the LOP DG start instrumentation channels are configured to provide a one out of three logic to initiate a trip of the incoming offsite power.~~

~~A Note is added to allow bypassing an inoperable channel for up to 4 hours for surveillance testing of other channels. This allowance is made where bypassing the channel does not cause an actuation and where at least two other channels are monitoring that parameter.~~

~~The specified Completion Time and time allowed for bypassing one channel are reasonable considering the Function remains fully OPERABLE on every bus and the low probability of an event occurring during these intervals.~~

B.1

~~Condition BA applies when one or more than one of the loss of voltage or more than one the degraded voltage channel functions (this includes both relays and timers) on a single bus are inoperable.~~

~~Required Action B.1 requires restoring all but one channel to OPERABLE status. The 1 hour Completion Time should allow ample time to repair most failures and takes into account the low probability of an event requiring an LOP start occurring during this interval.~~

(continued)

BASES

ACTIONS
(continued)

C-1

~~Condition C applies to each of the LOP DG start Functions when the Required Action and associated Completion Time for Condition A or B are not met.~~

In these circumstances the Conditions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, "AC Sources - Shutdown," for the DG made inoperable by failure of the LOP DG start instrumentation are required to be entered immediately. The actions of those LCOs provide for adequate compensatory actions to assure unit safety.

~~A Note is added to allow bypassing an inoperable channel for up to 2 hours for surveillance testing. This allowance is made where bypassing the channel does not cause an actuation and where at least one other channel is monitoring that parameter.~~

SURVEILLANCE
REQUIREMENTS

SR-3.3.5.1

~~Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.~~

~~Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.~~

~~The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.5.2

SR 3.3.5.2 is the performance of a TADOT. This test is performed every ~~{31 days}~~ 18 months. The test checks trip devices that provide actuation signals directly, bypassing the analog process control equipment. For these tests, the relay Trip Setpoints are verified and adjusted as necessary. The Frequency is based on the known reliability of the relays and controls and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.5.3

SR 3.3.5.3 is the performance of a CHANNEL CALIBRATION.

The setpoints, as well as the response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay, ~~as shown in Reference 1.~~

A CHANNEL CALIBRATION is performed every ~~{18} months, or~~ approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of ~~{18} months~~ is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an ~~{18} month~~ calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. FSAR, Section ~~{8.3}~~.
 2. FSAR, Chapter ~~{15}~~.
 3. ~~Unit Specific RTS/ESFAS Setpoint Methodology Study WCAP-11082 Rev. 2, Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Stations-Eagle 21 Version, May 1993.~~
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B 3.3 INSTRUMENTATION

B 3.3.6 Containment Purge and Exhaust ~~Ventilation~~ Isolation Instrumentation

BASES

BACKGROUND

Containment purge and exhaust ~~ventilation~~ isolation instrumentation closes the containment ~~ventilation~~ isolation valves in the Mini Purge System and the Shutdown Purge System. This action in conjunction with a Phase A signal isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Mini Purge or Vacuum/Pressure Relief System may be in use during reactor operation and the Shutdown Purge System will be in use with the OR reactor shutdown.

Containment purge and exhaust ~~ventilation~~ isolation initiates on a automatic safety injection (SI) signal through the Containment Isolation-Phase A Function, or by manual actuation of Phase A Isolation. The Bases for LCO.3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these modes of initiation.

Four ~~two~~ radiation monitoring channels are also provided as input to the containment purge and exhaust ~~ventilation~~ isolation. The ~~four~~ ~~two~~ channels measure containment radiation at ~~two~~ locations. One channel is a containment area gamma monitor, and the other three measure radiation in a sample of the containment purge exhaust. The three purge exhaust radiation detectors are of three different types: gaseous, particulate, and iodine monitors. All four in the exhaust duct for fan E-3. Both detectors will respond to most events that release radiation to containment. Both monitors are gaseous activity monitors that will respond to noble gases, particulate and iodine. The high alarm setpoint is based upon the design basis fuel handling accident source term which does not have a particulate component. The actual high alarm setpoint is more than a factor of 500 below the design calculation earliest actuation point. Since the monitors can only be adjusted to one high alarm setpoint and no particulate is expected during a fuel handling accident, a setpoint based on site boundary noble gases is conservative. However, analyses have not been conducted to demonstrate that all credible events will be detected by more than one monitor. Therefore, for the purposes of this LCO the four channels are not considered redundant. Instead, they are treated as four one out of one Functions. Since the purge exhaust monitors constitute a sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.

Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from any one ~~either~~ of the four ~~two~~ channels initiates containment

(continued)

BASES

BACKGROUND
(continued)

~~purge ventilation isolation, which closes both inner and outer the containment ventilation isolation valves in the Mini Purge System and the Shutdown Purge System. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."~~

APPLICABLE
SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event, within approximately 60 seconds. The isolation of the ~~purge containment ventilation~~ valves has not been analyzed mechanistically in the dose calculations, although its ~~rapid isolation, using a conservative isolation time~~ is assumed. The containment purge and exhaust ~~ventilation isolation~~ radiation monitors act as backup to the SI signal to ensure closing of the purge and exhaust ~~containment ventilation isolation~~ valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The containment purge and exhaust ~~ventilation~~ isolation instrumentation satisfies Criterion 3 of the ~~NRC Policy Statement 10 CFR 50.36(c)(2)(iii)~~.

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Purge and Exhaust ~~Ventilation~~ Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation
~~NOT USED~~

~~The LCO requires two channels OPERABLE. The operator can initiate Containment Purge Isolation at any time by using either of two switches in the control room. Either switch actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.~~

~~The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.~~

~~Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet.~~

(continued)

BASES

LCO
(continued)

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI, and ESFAS Function 3.a, Containment Phase A Isolation. The applicable MODES and specified conditions for the containment purge ventilation isolation portion of these Functions are different and less restrictive than those for their Phase A isolation and SI roles. If one or more of the SI or Phase A isolation Functions becomes inoperable in such a manner that only the Containment Purge Ventilation Isolation Function is affected, the Conditions applicable to their SI and Phase A isolation Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Purge Ventilation Isolation Functions specify sufficient compensatory measures for this case.

3. Containment Radiation

The LCO specifies ~~four~~ two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Purge Ventilation Isolation remains OPERABLE in MODES 1-4. ~~The LCO only requires one monitor to be OPERABLE during CORE ALTERATIONS or during movement of irradiated fuel assemblies in MODES 1-4. The LCO requires only one monitor to be operable during CORE ALTERATIONS or during movement of irradiated fuel~~

~~For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.~~

4. Containment Isolation - Phase A

Refer to LCO 3.3.2, Function 3.a., for all initiating Functions and requirements.

(continued)

BASES (continued)

APPLICABILITY The ~~Manual Initiation~~, Automatic Actuation Logic and Actuation Relays, Containment Isolation-Phase A, and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4, and during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment purge and exhaust ~~ventilation~~ isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, the containment purge and exhaust ~~ventilation~~ isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a ~~COTCFI and/or Channel Calibration~~, when the process instrumentation is set up for adjustment to bring it within specification. ~~Drift can also be observed during a Channel check or CFI and if observed would prompt action to correct the discrepancy.~~ If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment purge ~~ventilation~~ isolation radiation monitor channel. ~~Since the four containment radiation monitors measure different parameters,~~

(continued)

BASES

ACTIONS

A.1 (continued)

~~failure of a single channel may result in loss of the radiation monitoring function for certain events. Consequently, the failed channel must be restored to OPERABLE status. The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.~~

~~A Note has been added to state that Condition A is only applicable in MODE 1, 2, 3, or 4.~~

B.1

Condition B applies to all Containment Purge and Exhaust Ventilation Isolation Functions and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for these Functions. It also addresses the failure of ~~multiple~~ both radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, ~~multiple~~ both radiation channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

C.1 and C.2

Condition C applies to all Containment Purge and Exhaust Ventilation Isolation Functions and addresses the train orientation of the SSPS and the master and slave relays for these Functions. It also addresses the ~~failure of multiple~~ condition of no OPERABLE radiation monitoring channels, ~~or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.~~ If a train is inoperable, ~~multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met~~ required radiation monitor is inoperable, operation may continue as long as the Required Action to place and maintain containment purge and exhaust Ventilation isolation

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

valves (RCV-11, 12, FGV 660, 661, 662, 663, 664) in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Purge and Exhaust Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.1 (continued)

channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

SR 3.3.6.3

SR 3.3.6.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

SR 3.3.6.4

A ~~GO/GET~~ is performed every 92 days on each required channel to ensure the entire channel will perform the intended function. The frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). This test verifies the capability of the instrumentation to provide the containment purge and exhaust system isolation. The

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.4 (continued)

~~setpoint shall be left consistent with the current unit specific calibration procedure tolerance.~~

SR 3.3.6.5

SR 3.3.6.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every ~~[92]~~ days. The Frequency is acceptable based on instrument reliability and industry operating experience.

SR 3.3.6.6

~~There is no manual actuation of CVI except via phase A or B. This testing is performed as part of 3.3.2.~~

~~SR 3.3.6.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every [18] months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).~~

~~The test also includes trip devices that provide actuation signals directly to the SSPS, bypassing the analog process control equipment. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.~~

~~The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.~~

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.6.7

A CHANNEL CALIBRATION is performed every [18] months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

SR 3.3.6.8

This SR assures that the individual channel RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the FSAR. Individual component response times are not modeled in the analyses. The analyses model the overall or elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., valves in full closed position). The response time may be measured by a series of overlapping tests such that the entire response time is measured.

RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 18 months. The 18 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. 10 CFR 100.11.
2. NUREG-1366, December 1992.

B 3.3 INSTRUMENTATION

B 3.3.7 Control Room Emergency Filtration Ventilation System (GREFS CRVS) Actuation Instrumentation

BASES

BACKGROUND

The GREFS CRVS provides an enclosed control room environment from which the both units can be operated following an uncontrolled release of radioactivity. During normal operation, the Auxiliary Building Ventilation System provides control room ventilation. Upon receipt of an actuation signal, the GREFS CRVS shifts from normal operation and initiates filtered ventilation and pressurization of the control room. This system is described in the Bases for LCO 3.7.10, "Control Room Emergency Filtration Ventilation System," and is common to both units.

The actuation instrumentation consists of redundant radiation monitors in the air intakes and to the control room areas. There are two detectors in each of the two normal control room air intakes. However, since they take suction from a common area, the North and South sides of the mechanical equipment room, only two detectors are required to provide protection against a single failure. A Phase "A" containment isolation signal or a high radiation signal from any either of these required detectors in the normal intake will initiate both trains of the GREFS CRVS pressurization from the pressurization intake with the lowest radiation level (each pressurization intake, one on the North end of the turbine building and one on the South, has two radiation monitors). The control room operator can also initiate GREFS CRVS pressurization trains by manual switches in the control room. The GREFS is also actuated by a safety injection (SI) signal. The SI Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

The CRVS has two additional manually selected operating modes, smoke removal and recirculation. Neither of modes are required for the CRVS to be OPERABLE but they are useful for certain non-DBA circumstances.

APPLICABLE SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The GREFS CRVS acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

(continued)

BASES

In MODES 1, 2, 3, and 4, the radiation monitor actuation of the CREFS CRVS is a backup for the SI Phase A signal actuation. This ensures initiation of the CREFS CRVS during a loss of coolant accident or steam generator tube rupture involving a release of radioactive materials.

The radiation monitor actuation of the CREFS in MODES 5 and 6, during movement of irradiated fuel assemblies E and

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

~~CORE ALTERATIONS~~, is the primary means to ensure control room habitability in the event of a fuel handling or waste gas decay tank rupture accident. The GREFS CRVS ~~pressurization system~~ actuation instrumentation satisfies Criterion 3 of the NRC Policy Statement 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that instrumentation necessary to initiate the GREFS ~~CRVS pressurization system~~ is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the GREFS ~~CRVS pressurization mode~~ at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

~~Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet.~~

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Actuation Logic and Relays OPERABLE to ensure that no single random failure can prevent automatic actuation ~~of the pressurization system~~.

~~Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b., SI, in LCO 3.3.2. The applicable MODES and specified conditions for the GREFS portion of these functions are different and less restrictive than those specified for their SI roles. If one or more of the SI functions becomes inoperable in such a manner that only the GREFS function is affected, the Conditions applicable to their SI function need not be entered. The less~~

(continued)

BASES

LCO

2. Automatic Actuation Logic and Actuation Relays (continued)

~~restrictive Actions specified for inoperability of the CREFS Functions specify sufficient compensatory measures for this case.~~

3. Control Room Radiation

The LCO specifies two required Control Room Atmosphere ~~Normal Intake Radiation Monitors~~ and ~~two required Control Room Air Intake Radiation Monitors~~ at each to ensure that the radiation monitoring instrumentation necessary to initiate the CREFS ~~CRVS~~ ~~pressurization system~~ remains OPERABLE.

~~For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.~~

4. Safety Injection

~~Refer to LCO 3.3.2. Function 1, for all initiating Functions and requirements.~~

APPLICABILITY

The CREFS ~~CRVS~~ Functions must be OPERABLE in MODES 1, 2, 3, 4, ~~and during CORE ALTERATIONS~~ and movement of irradiated fuel assemblies. The Functions must also be OPERABLE in MODES ~~5 and 6~~ when required for a waste gas decay tank rupture accident, to ensure a habitable environment for the control room operators.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather

(continued)

BASES

ACTIONS
(continued)

than a total loss of function. This determination is generally made during the performance of a GOTCFI and/or Channel Calibration, when the process instrumentation is set up for adjustment to bring it within specification. Drift can also be observed during a Channel check or CFI and if observed would prompt action to correct the discrepancy. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.7-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the actuation logic train Function of the CREFS CRVS, the radiation monitor channel Functions, and the manual channel Functions.

If one train is inoperable, or one radiation monitor channel is inoperable in one or more Functions, 7 days are permitted to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this Completion Time is the same as provided in LCO 3.7.10. If the channel/train cannot be restored to OPERABLE status, one CREFS CRVS train must be placed in the emergency radiation protection pressurization mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

~~The Required Action for Condition A is modified by a Note that requires placing one CREFS train in the toxic gas protection mode instead of the [radiation protection] mode of operation if the automatic transfer to toxic gas protection mode is inoperable. This ensures the CREFS train is placed in the most conservative mode of operation relative to the OPERABILITY of the associated actuation instrumentation.~~

(continued)

BASES

ACTIONS
 (continued)

B.1.1, B.1.2, and B.2

Condition B applies to the failure of two CREFS ~~CRVS~~ actuation trains, two radiation monitor channels, or two manual channels. The first Required Action is to place one CREFS ~~CRVS~~ train in the emergency ~~[radiation protection]~~ ~~pressurization~~ mode of operation immediately. This accomplishes the actuation instrumentation Function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.10 must also be entered for the CREFS ~~CRVS~~ train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.10.

Alternatively, both trains may be placed in the emergency ~~[radiation protection]~~ ~~pressurization~~ mode. This ensures the CREFS function is performed even in the presence of a single failure.

~~The Required Action for Condition B is modified by a Note that requires placing one CREFS train in the toxic gas protection mode instead of the [radiation protection] mode of operation if the automatic transfer to toxic gas protection mode is inoperable. This ensures the CREFS train is placed in the most conservative mode of operation relative to the OPERABILITY of the associated actuation instrumentation.~~

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS
(continued)

D.1 and D.2

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met ~~during CORE ALTERATIONS or~~ when irradiated fuel assemblies are being moved. Movement of irradiated fuel assemblies ~~and CORE ALTERATIONS~~ must be suspended immediately to reduce the risk of accidents that would require CREFS actuation.

E.1

Condition E applies when the Required Action and associated Completion Time for Condition A or B have not been met in MODE 5 or 6. Actions must be initiated to restore the inoperable train(s) to OPERABLE status immediately to ensure adequate isolation capability in the event of a waste gas decay tank rupture.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CREFS Actuation Functions.

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.7.1 (continued)

including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.7.2

A ~~COT CFT~~ is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the CREFS ~~CRVS~~ actuation. ~~The setpoints shall be left consistent with the unit specific calibration procedure tolerance.~~ The Frequency is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.7.3

SR 3.3.7.3 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 31 days on a STAGGERED TEST BASIS. The Frequency is justified in WCAP-10271-P-A, Supplement 2, Rev. 1 (Ref. 1).

SR 3.3.7.4

SR 3.3.7.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.7.4 (continued)

check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The Frequency is acceptable based on instrument reliability and industry operating experience.

SR 3.3.7.5

SR 3.3.7.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every ~~92] days~~ 18 months. The Frequency is acceptable based on instrument reliability and industry operating experience (Ref 1 and 2).

SR 3.3.7.6

SR 3.3.7.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every ~~18] months~~. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The test also includes trip devices that provide actuation signals directly to the Solid State Protection System, bypassing the analog process control equipment. The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience. The SR is modified by a Note that excludes verification of setpoints during the

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.7.6 (continued)

TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.7.7

A CHANNEL CALIBRATION is performed every ~~18~~ months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. WCAP-13878, "reliability of Potter & Brumfield MDR Relays", June 1994
 2. WCAP-13900, "Extension of Slave Relay Surveillance Test intervals", April 1994
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B 3.3 INSTRUMENTATION

B 3.3.8 Fuel Handling Building Air Cleanup Ventilation System (FBACSFHBVS) Actuation Instrumentation

BASES

BACKGROUND

The FBACSFHBVS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident or a loss of coolant accident (LOCA) are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.13, "Fuel Handling Building Air Cleanup Ventilation System." The system initiates filtered ventilation of the fuel building automatically following receipt of a high radiation signal (gaseous or particulate) or a safety injection (SI) signal from the Spent Fuel Pool Monitor or from the New Fuel Storage Vault Monitor (or from gaseous monitors 45 A/B when installed). Initiation may also be performed manually as needed from the main control room or fuel handling building.

High gaseous and particulate radiation, each monitored by from either of the two monitors, provides FBACSFHBVS initiation. Each FBACS train is initiated by high radiation detected by a channel dedicated to that train. There are a total of two channels, one for each train. Each channel contains a gaseous and particulate monitor. High radiation detected by any monitor or an SI signal from the Engineered Safety Features Actuation System (ESFAS) initiates fuel building isolation and starts the FBACS. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the fuel building. Since the radiation monitors include an air sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.

APPLICABLE SAFETY ANALYSES

The FBACSFHBVS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident or a LOCA are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the fuel building exhaust following a LOCA or fuel handling accident so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1).

The FBACSFHBVS actuation instrumentation satisfies Criterion 3 of the NRC Policy Statement 10 CFR 50.36(c)(2)(iii).

(continued)

BASES (continued)

LCO The LCO requirements ensure that instrumentation necessary to initiate the FBACSFHBVS is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the FBACSFHBVS at any time by using either of two switches: one in the control room and another in the fuel handling building. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

~~Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet.~~

~~2. Automatic Actuation Logic and Actuation Relays~~

~~The LCO requires two trains of Actuation Logic and Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.~~

~~Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b., SI, in LCO 3.3.2. The applicable MODES and specified conditions for the FBACS portion of these functions are different and less restrictive than those specified for their SI roles. If one or more of the SI functions becomes inoperable in such a manner that only the FBACS function is affected, the Conditions applicable to their SI function need not be entered. The less restrictive Actions specified for inoperability of the FBACS functions specify sufficient compensatory measures for this case.~~

3 ~~2~~. Fuel Building Radiation

The LCO specifies two required Gaseous Radiation Monitor channels and two required ~~Particulate Radiation Monitor~~ channels to ensure that the radiation monitoring instrumentation necessary to initiate the FBACS FBVS remains OPERABLE.

(continued)

BASES

LCO 3 2. Fuel Building Radiation (continued)

~~For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, filter motor operation, detector OPERABILITY, if these supporting features are necessary for actuation to occur under the conditions assumed by the safety analyses.~~

Only the Trip Setpoint is specified for each FBACSFHBVS Function in the LCO. The Trip Setpoint limits account for instrument uncertainties, which are defined in the Unit Specific Setpoint Calibration Procedure (Ref. 2).

APPLICABILITY The manual FBACS FBVS initiation must be OPERABLE in ~~MODES [1, 2, 3, and 4]~~ and when moving irradiated fuel assemblies in the fuel building, to ensure the FBACSFHBVS operates to remove fission products associated with leakage after a LOCA or a fuel handling accident. ~~The automatic FBACS actuation instrumentation is also required in MODES [1, 2, 3, and 4] to remove fission products caused by post LOCA Emergency Core Cooling Systems leakage.~~

High radiation initiation of the FBACSFHBVS must be OPERABLE in any MODE during movement of irradiated fuel assemblies in the fuel building to ensure automatic initiation of the FBACS FBVS when the potential for a fuel handling accident exists.

While in MODES 5 and 6 without fuel handling in progress, the FBACSFHBVS instrumentation need not be OPERABLE since a fuel handling accident cannot occur.

ACTIONS The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a ~~GOTCFI and/or Channel Calibration~~, when the process instrumentation is set up for adjustment to bring it within

(continued)

BASES

ACTIONS
(continued)

specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered. Drift can also be observed during a Channel check or CFI and if observed would prompt action to correct the discrepancy.

Two Notes have been added to the ACTIONS to clarify the application of Completion Time rules and the Applicability of LCO 3.0.3. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4 the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in Mode 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1

Condition A applies to the actuation logic train function of the Solid State Protection System (SSPS), the radiation monitor functions, and the manual function. Condition A applies to the failure of a single actuation logic train, one or more radiation monitor channels, or manual channels. If one or more channels or trains are inoperable, a period of 730 days is allowed to restore it to OPERABLE status. If the train cannot be restored to OPERABLE status, an appropriate portable continuous monitor with the same setpoint, or an individual qualified in radiation protection procedures with a dose rate monitoring device must be in the spent fuel pool area, one FBACSFHBVS train must be placed in the Iodine Removal mode of operation immediately. This effectively accomplishes the actuation instrumentation function and places the unit area in a conservative mode of operation or provides appropriate monitoring for continued fuel movement. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this time is the same as that provided in LCO 3.7.13.

B.1.1, B.1.2, B.2

Condition B applies to the failure of two FBACS actuation logic trains, two radiation monitors, or two manual channels. The Required Action is to place one FBACS train in operation

(continued)

BASES

~~immediately. This accomplishes the actuation instrumentation function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.13 must also be entered for the FBAGS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed on train inoperability as discussed in the Bases for LCO 3.7.13.~~

(continued)

BASES

ACTIONS B.1.1, B.1.2, B.2 (continued)

~~Alternatively, both trains may be placed in the emergency [radiation protection] mode. This ensures the FBACS Function is performed even in the presence of a single failure.~~

C.1

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and irradiated fuel assemblies are being moved in the fuel building. Movement of irradiated fuel assemblies in the fuel building must be suspended immediately to eliminate the potential for events that could require FBACS FBVS actuation.

D.1 and D.2

~~Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.~~

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which SRs apply to which FBACSFHBVS Actuation Functions.

~~Notes have been added that clarify which functions will be associated with which monitors when the new radiation monitors RM 45A and 45B are installed.~~

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.8.1 (continued)

channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.8.2

A ~~GOT~~ ~~CFM~~ is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the FBACS actuation. ~~The setpoints shall be left consistent with the unit specific calibration procedure tolerance.~~ The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.8.3

~~SR 3.3.8.3 is the performance of an ACTUATION LOGIC TEST. The actuation logic is tested every 31 days on a STAGGERED TEST BASIS. All possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency is based on the known~~

(continued)

BASES
SURVEILLANCE
REQUIREMENTS

~~SR 3.3.8.3 (continued)~~

~~reliability of the relays and controls and the multichannel
redundancy available, and has been shown to be acceptable through
operating experience.~~

SR 3.3.8.4

SR 3.3.8.4 is the performance of a TADOT. This test is a check of the manual actuation functions and is performed every ~~[18]~~ months. Each manual actuation function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Frequency is based on operating experience and is consistent with the typical industry refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.8.5

A CHANNEL CALIBRATION is performed every ~~[18]~~ months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.
 2. ~~Unit Specific Setpoint Calibration Procedure.~~
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Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck out" and the correct material is inserted using the "redline" feature. If the "generic" is correct, the information is "redlined". The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Methodology For Mark-up of NUREG-1431 Bases
(continued)

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the markup process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(10 Pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.3

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>JUSTIFICATION</u>
3.3-01	This trip function or design feature is not included in the plant design or it is not credited and has no safety function.
3.3-02	For the Reactor Trip on Turbine Trip function based on turbine stop valve position, 4 of 4 channels are required to close to less than 1% open in order to generate the reactor trip signal. Thus, it is acceptable to have more than one Turbine Stop Valve Closure - reactor trip function channel inoperable and placed in trip per current TS Table 3.3-1, Functional Unit [17.b], ACTION Statement [7]. In addition, the 4 hour bypass note applies only to the [Low Auto Stop Oil Pressure] channels. ITS 3.3.1 Condition P has been revised.
3.3-03	This change to ITS 3.3.1 Condition R is consistent with the current licensing basis. A 4-hour AOT for SSPS logic surveillance testing has little usefulness if the RTBs cannot be bypassed for the duration of that testing. RTB surveillance testing retains the current 2-hour AOT.
3.3-04	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 6B).
3.3-05	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.3-06	Retains CTS power requirement of 75% RTP in the ITS SR 3.3.1.6 Note concerning when the incore/excore calibration is performed. The ISTS proposal would require unnecessary delays in the post-refuel power ascension. As per the current TS 4.0.4 exception, it is acceptable to go above 75% RTP during power ascension provided the calibration is performed within 24 hours of exceeding 75% RTP. [The Note is further revised to permit achieving equilibrium conditions (per CTS 4.2.2.2.d.1) prior to performing the required surveillance.]
3.3-07	Note 3 is added to ITS SR 3.3.1.11 to be consistent with the CTS Table 4.3-1 Note [5]. This ensures that this exception for power and intermediate range detector plateau voltage verification, as discussed in the ITS BASES for SR 3.3.1.11, is included in the Technical Specifications rather than being only found in the BASES. The note replaces the exception to LCO 3.0.4 in the current TS.
3.3-08	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.3-09	The addition of footnote [(m)] to ITS Table 3.3.1-1 for Function 10 clarifies the low flow setpoint relationship to the quantity identified as Minimum Measured Flow, consistent with the current TS.

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- 3.3-10 [The Overtemperature ΔT setpoint equation had a bracket in the wrong place and was corrected. The $OT_{\Delta T}$ equation in NUREG-1431 Note 1 shows the lead-lag compensation associated with the term $(1 + \tau_1 S)/(1 + \tau_2 S)$ (note that CN 3.3-13 revised the tau subscripts per CTS) to be applicable to the T' term. This is incorrect. This error can also be found in Revision 4a of NUREG-0452. This error apparently was taken from Section A-3 and Figure A-1 of WCAP-8745-P and WCAP-8746 as well as from Figure 6.1-4 of WCAP-7907-P-A, "LOFTRAN Code Description," April 1984. The latter reference has been corrected in Figure 6.1-4 of WCAP-7878, "LOFTRAN Code Description and User's Manual," Revision 5, November 1989. The lead-lag compensation applies only to the measured T_{avg} . This is consistent with the manner in which the electronics have always processed the $OT_{\Delta T}$ setpoint signal, as depicted on Westinghouse drawing 8756D37 sheets 7-10 described in FSAR Section 7.2]. Another change needed for the Overtemperature ΔT setpoint equation concerns the inequality sign for the K_2 term. As defined in NUREG-1431, this term has a " \geq " sign. In this case, the Overtemperature ΔT setpoint would be conservatively decreased if T_{avg} were increasing above [576.6°F for Unit 1 and 577.6°F for Unit 2], i.e. with $(T-T')$ a positive value. However, if T_{avg} were decreasing below [576.6°F or 577.6°F, for the respective Units,] such that $(T-T')$ is a negative value, the \geq sign could result in an unwanted increase in the Overtemperature ΔT setpoint. Therefore, the inequality sign for K_2 is changed to an equal sign, consistent with the current TS. This issue is avoided in the construct of the Overtemperature ΔT setpoint by setting K_5 and K_6 to zero for decreasing T_{avg} , i.e. K_5 and K_6 are conditionally defined. In addition, the f_1 (ΔI) penalty function was corrected to reflect correct decimal point placement and to ensure a reduction in the setpoint if $(q_i - q_b)$ is outside the deadband. The f_1 (ΔI) value must be positive such that it lowers the setpoint when subtracted. The inequality signs around the deadband were corrected to reflect a zero penalty when $(q_i - q_b)$ is within the deadband. Decimal point placement corrections have been made to recognize that the penalty function gains have units of (°F or % ΔT span per % RTP).
- 3.3-11 Added "or Rod Control System incapable of rod withdrawal," which makes Note (f) the complete antithesis of Note (b).
- 3.3-12 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-13 The equations for Overtemperature ΔT and Overpower ΔT are revised to be consistent with the CTS. The value of the time constant τ_6 has always been 0 seconds and the factor utilizing this time constant has not been shown as part of the Overtemperature ΔT equation in licensing documents since the factor value has been unity. Thus, the factor utilizing this time constant has been deleted.
- 3.3-14 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-15 The CHANNEL CHECK surveillance (SR 3.3.2.1) is deleted from the P-11 [] interlock because CHANNEL CHECKS are not applicable for permissive functions. This change is consistent with the current TS.
- 3.3-16 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-17 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-18 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-19 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).

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NUMBER

JUSTIFICATION

- 3.3-20 This change adds note 2 on [Containment Radiation Level (High Range)] calibration in ITS SR [3.3.3.2] to be consistent with current TS Table [4.3-7 Note (2)]. This note is acceptable as it reflects the unique calibration requirements of these high range radiation monitors as defined in the current TS.
- 3.3-21 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-22 Consistent with CTS [3.3.3.5], [RCP breaker indication is excluded from CHANNEL CHECKS and reactor trip breaker and RCP breaker indications are excluded from] CHANNEL CALIBRATIONS in ITS SR 3.3.4.3 since these SRs have no meaning for [these] functions.
- 3.3-23 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-24 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-25 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-26 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-27 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-28 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-29 {Not used}
- 3.3-30 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-31 The current TS require the response times associated with the [undervoltage and degraded voltage diesel generator start functions and the] containment [purge and exhaust] isolation functions to be verified against the specific response time values contained in the [FSAR]. The ITS is revised to match the current TS and the response time values are [moved to the FSAR per CN 01-35-LG]. As is done with the Reactor Trip System and the ESFAS instrumentation, this method is an appropriate way to control response times. [SR 3.3.5.4 and SR 3.3.6.8] are added to require the response time verifications.
- 3.3-32 Improved TS [3.3.6 ACTION A is modified by a Note and] Table 3.3.6-1 is changed to be consistent with current TS [3.3.2 Functional Unit 3.c and current TS 3.9.9]. Subfunctions [b, c and d] of Containment Radiation are stricken since only the gaseous [] channel provides the actuation function [and the bracketed setpoint is changed to reflect plant-specific requirements]. [The number of gaseous monitors required for CORE ALTERATIONS or during movement of irradiated fuel has been revised to one (either RM 44A or B) per the CTS].
- 3.3-33 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-34 This change adds an LCO 3.0.3 exception Note 1 to ITS 3.3.8 to reflect industry Traveler TSTF-36, Rev. 2.
- 3.3-35 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-36 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

**CHANGE
NUMBER**

JUSTIFICATION

- 3.3-37 Several ITS Required Action Notes are modified per CTS to allow a channel to be placed in bypass for surveillance testing. []
- 3.3-38 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-39 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-40 This change adds "and setpoint adjustment" to ITS 3.3.1 Condition E, similar to the Note for Condition D. Setpoint adjustment is required by the Required Actions of other specifications. The clarity and consistency of the specification is enhanced by adding this note to Condition E, in the same manner as Condition D.
- 3.3-41 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-42 This change deletes ITS 3.3.1 Condition N per Traveler TSTF-169. Condition M is appropriate for Function 10.a to prevent sequential entry into Condition N followed by M and exceeding the evaluated Completion Time in WCAP-10271-P-A, Supplement 2, Rev. 1. With this change, there is no need to list separate Functions 10.a and 10.b and combining the Functions eliminates Applicability questions similar to the Condition M vs. N concern above.
- 3.3-43 This change revises ITS 3.3.1 Condition R Notes 1 and 2 per Traveler TSTF-168. The 2-hour AOT should not be limited to only UVTA/STA maintenance. This is consistent with the current TS and is acceptable because the specific maintenance activity which requires that a reactor trip breaker be bypassed does not affect the impact of having the breaker bypassed. []
- 3.3-44 This change revises ITS 3.3.1 Conditions S and T and ITS 3.3.2 Condition L, as well as the number of Required Channels in Tables 3.3.1-1 and 3.3.2-1, to reflect current TS ACTION Statements [8 and 21]. The Conditions apply to one or more channels [or trains, as Condition T applies to permissive P-7,] because the safety function is served with the interlock in the appropriate state for existing plant conditions. The existing plant design only requires 3 of the 4 channels (2-out-of-3 for P-11) for these interlocks to be operable.
- 3.3-45 A new CONDITION and SR are added for the current licensing basis required seismic trip.
- 3.3-46 A new CONDITION and SR are added for the current licensing basis required Steam Generator level low-low time delay trip. These changes affect both ITS 3.3.1 and 3.3.2.
- 3.3-47 Note 2 of SR 3.3.1.2 is revised to limit the power increase to less than 30% per the current licensing basis before the SR is complete.
- 3.3-48 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-49 ITS SR 3.3.1.8 is revised to extend the conditional COT frequency for power and intermediate range channels from 4 hours after reducing power below P-10 to 12 hours, based on operating experience regarding the time needed to perform the COTs. It stands to reason that if 4 hours are allowed for 2 Source Range COTs, 12 hours should be allowed for 6 Intermediate Range and Power Range COTs. The SR continues to assure that the COTs are performed in a timely manner after the requisite plant conditions are entered. This change is consistent with Traveler WOG-106.
- 3.3-50 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

CHANGE
NUMBER

JUSTIFICATION

- 3.3-51 ITS ACTION B.2 of LCO 3.3.7 is deleted, since DCPD cannot operate with both pressurization systems running at the same time. The design of the system is such that operation of two pressurization fans would over pressurize the supply ducting to the filters.
- 3.3-52 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.3-53 The REQUIRED CHANNELS description for Functions 2.a and 3.b.(1), of ITS Table 3.3.2-1, are revised per the CTS to note that only two switches (one per train) exist and that both must be moved coincident for manual initiation.
- 3.3-54 Function 18.b (P-7) of ITS Table 3.3.1-1 is clarified. COTs and Channel Calibrations apply to the P-10 and P-13 inputs, not to the P-7 logic function. This change is an administrative clarification to address the relationships between these interlocks in the plant's design.
- 3.3-55 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.3-56 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.3-57 Not used.
- 3.3-58 This change adds new ITS 3.3.2 Condition [N] to reflect current TS Table 3.3-3 ACTION Statement [24] on manual AFW [and manual MSIV closure] initiation.
- 3.3-59 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.3-60 Consistent with the design and current TS, Surveillance Requirements 3.3.2.3 and 3.3.2.7 are not used by any function listed in Table 3.3.2-1 and are deleted.
- 3.3-61 This change revises the ITS SR 3.3.2.11 Frequency to 18 months per current TS Table 4.3-2 Functional Unit [8.c], which is the ESFAS P-4 permissive. The 18 month Frequency for the surveillance of the basic switch logic associated with the opening of the reactor trip breakers is the value specified in the current TS. [Deleted the Note stating that verification of set point is not required per the CTS.]
- 3.3-62 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.3-63 This change revises ITS Table 3.3.2-1 [Notes (b) and (g)] per current TS Table [3.3-3] Notes [# and ##]. This revision is a clarification to the operator that describes the circumstances under which the [Steamline Pressure Negative Rate - High, Steam Pressure-low, or Pressurizer Pressure-low functions may be or are blocked relative to the] P-11 permissive.
- 3.3-64 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.3-65 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.3-66 The MODE 4 requirement of the CTS is retained and added to Table 3.3.2-1 for SI actuated by Containment Pressure high-high. ITS 3.3.2 ACTIONS D and E are revised accordingly.
- 3.3-67 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).

CHANGE
NUMBER

JUSTIFICATION

- 3.3-68 A Note is added to state that CONDITION D is only applicable in MODES 1 and 2. A new CONDITION H is added to require entering MODE 3 if CONDITION B is not met when entered due to not meeting CONDITION D. These changes are per the CTS.
- 3.3-69 {The phrase "...that is not normally energized" is deleted per the CTS. All of the instrumentation listed is normally energized at power.}
- 3.3-70 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-71 This change revises Table 3.3.3-1 per the reviewers note to update CTS PAM instruments per the requirements of Reg. Guide 1.97.
- 3.3-72 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-73 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-74 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-75 The CHANNEL FUNCTIONAL TEST is substituted for the COT per current licensing basis.
- 3.3-76 Consistent with the current design and TS, a Trip Actuating Device Operational Test (TADOT) is not required for any of the functions explicitly listed in Table 3.3.6-1; therefore, the associated Surveillance Requirement is deleted. Note that a TADOT is required in accordance with LCO 3.3.2 for functions 3.a.1 and 2.a, as referenced in the Table.
- 3.3-77 Containment Vent Isolation is initiated by the ESFAS Phase "A" isolation signals. As such, the number of required channels and required surveillances for the manual initiation of Containment Vent Isolation are captured by the requirements for Phase "A" isolation in the ESFAS tables.
- 3.3-78 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-79 This change adds APPLICABILITY columns to ITS Tables 3.3.6-1 and 3.3.7-1 to reflect current TS with varying Functional Applicabilities. This change is consistent with the format used for the RTS and ESFAS instrumentation in the ITS and is a clearer method to present varying Applicabilities from the current TS. These changes are administrative format changes that insert the Applicabilities from the current TS into Tables 3.3.6-1 and 3.3.7-1. This change is consistent with traveler TSTF-161.
- 3.3-80 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-81 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-82 The CONDITIONS, REQUIRED ACTIONS, etc. are revised per the current licensing basis. The plant FBACS does not perform any accident mitigation functions except during the fuel handling accident
- 3.3-83 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-84 The Note of SR 3.3.4.4 is deleted since Table 3.3.4-1 does not have a neutron detector specified per the current TS.
- 3.3-85 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).

CHANGE
NUMBER

JUSTIFICATION

- 3.3-86 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-87 Not used.
- 3.3-88 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-89 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-90 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-91 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-92 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-93 ITS 3.3.1 Condition V is deleted. It is not entered from Table 3.3.1-1 nor do the Bases clarify when it would be needed, raising the concern of misinterpretation. Condition V does not replace LCO 3.0.3 requirements to assess when the plant is outside the licensing basis. There is no similar ACTION Statement in the current TS for the Reactor Trip System. This change is consistent with Traveler TSTF-135.
- 3.3-94 ITS 3.3.4 is revised per current TS [3.3.3.5] with regard to [remote shutdown panel] controls. [Remote shutdown panel] controls are added to the LCO, Condition A, and SR 3.3.4.2. By explicitly including the controls, the specification is clarified to be more than instrumentation. This change is acceptable because it does not change the meaning while retaining the clarity of the CTS.
- 3.3-95 ITS 3.3.1 Condition H, Required ACTION H.1, and the second part of Function 4 Applicability (MODE 2 below P-6) in ITS 3.3.1 are deleted since they provide no real compensatory measures. With their deletion, there is no need to repeat the > P-6 Applicability in Conditions F and G. In accordance with LCO 3.0.4, the intermediate range detectors must be OPERABLE prior to entering the Applicability of the retained part of Function 4 (i.e., MODE 2 above P-6). Condition H and Required ACTION H.1 ensure the same thing and, therefore, can be deleted. This change is consistent with Traveler TSTF-135.
- 3.3-96 [] Note 2 for ITS SR 3.3.1.3 is revised to replace the bracketed 15% RTP power level constraint with 50% RTP. The specified power level in ITS SR 3.3.1.3 should reflect the applicable safety analysis basis consistent with the [APPLICABILITY and] Required Actions of ITS LCO 3.2.3 (AFD) and LCO 3.2.4 (QPTR).
- As revised, this surveillance requirement is acceptable in that it assures the surveillance is performed after the appropriate plant conditions are attained and still provides sufficient time to perform the surveillance in a controlled manner.
- 3.3-97 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-98 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-99 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).

CHANGE
NUMBER

JUSTIFICATION

- 3.3-100 Not used.
- 3.3-101 The Note for ITS SR 3.3.1.12 is deleted since the plant design no longer includes the RTD bypass. The SR is retained and is applied to the required seismic trip instrumentation per the current licensing basis. Where cited in Table 3.3.2-1, a change to SR 3.3.1.10 has been made.
- 3.3-102 The control room (CR) does not have CR Atmosphere monitors as part of its current design. There are redundant CR intake monitors for each intake. The normal control room intakes are in an area common to both units, thus there are a total of four normal intake monitors. However, only two monitors, one from each unit, are required for the CRVS to be OPERABLE; this is explained in the Bases.
- 3.3-103 Function 11 of ITS Table 3.3.1-1 is revised per the CTS to reflect the current plant design of only a two loop trip. With this revision, ACTION O is no longer used since it was only applicable to the single loop trip.
- 3.3-104 CONDITION A of ITS 3.3.5 is revised to incorporate CTS ACTION-16. CONDITIONS B and C are not used.
- 3.3-105 Function 4.d.(2) of ITS Table 3.3.2-1 and Notes (c) and (h) are revised per current licensing basis.
- 3.3-106 ISTS 3.3.1 Required Actions B.2.2 and U.2.2 are not used, consistent with the current TS requirements of LCO 3.3.1 ACTION Statements [1, and 12] and the Applicability for ITS Table 3.3.1-1 Functions 1 and 20. The current TS provide for these Functions to be restored to OPERABLE status within 48 hours or the plant must be in HOT STANDBY within the next 6 hours when the plant is initially in MODES 1 or 2. Once HOT STANDBY has been reached, the shutdown mode applicabilities, i.e. MODES 3, 4, and 5, prevail. When in these MODES, another 48 hour AOT is allowed by the current TS or rod withdrawal must be precluded in the next one hour. Therefore, ISTS Required Actions B.2.2 and U.2.2 for Functions 1 and 20 in Table 3.3.1-1 are not necessary since the performance of Required Action B.2.1 and U.2.1 takes the plant to MODE 3, exits the Applicability, and requires entry into Condition C. This change is consistent with ITS 1.3 and 3.0.4.
- 3.3-107 Based upon operating experience to change Thermal Power in a controlled fashion without challenging the plant and consistent with the current TS which does not have a Completion Time for restoring one channel to OPERABLE status; but does prevent going above P-10 until it is restored, the Completion Time for ITS 3.3.1 Required Actions F.1 and F.2 should be increased to 24 hours. Condition F of ITS 3.3.1 is for one Intermediate Range Neutron Flux channel inoperable. Reactor protection would be provided by the OPERABLE Intermediate Range Neutron Flux channel and OPERABLE Power Range Neutron Flux channels. Indication would be available from the OPERABLE Intermediate Range Neutron Flux channel [, from OPERABLE Gamma-Metrics Neutron Flux detectors,] and from OPERABLE Power Range Neutron Flux channels with power approaching P-10. The Westinghouse Owners Group is considering a generic change on this issue, but deliberations were not completed at the time of our submittal.
- 3.3-108 Not used..

CHANGE
NUMBER

JUSTIFICATION

- 3.3-109 Not used.
- 3.3-110 Not used.
- 3.3-111 This change adds a Note to ITS SR 3.3.1.7 for source range instrumentation to verify interlocks P-6 and P-10 are in their required state for existing unit conditions. This is consistent with the current TS and is an enhancement which is easily performed and provides additional assurance that the interlocks are functioning correctly.
- 3.3-112 Not used.
- 3.3-113 Not used.
- 3.3-114 Not used.
- 3.3-115 Not used.
- 3.3-116 ACTION J of ITS 3.3.2 is not used since DCPD does not rely on motor-driven AFW pump start with loss of both main FW pumps. The function exists, but is not credited in any accident analysis and is not part of ESFAS Function 6 in the CTS.
- 3.3-117 This change to ITS 3.3.1 Condition R reflects current TS Table [3.3-1, ACTION Statement 12] which was based on NRC Generic Letter 85-09.
- 3.3-118 This change is for consistency with ITS 3.7.10 Condition [G].
- 3.3-119 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.3-120 ITS 3.3.1 Condition D is revised to reflect ITS SR 3.2.4.2 and CN 3.2-15 in the 3/4.2 package.
- 3.3-121 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.3-122 ITS 3.3.1 APPLICABILITY Note (b) for Functions 1, 5, 19-21 and Conditions C and K are revised to replace ACTIONS requiring the RTBs to be opened with ACTIONS that ensure subcriticality is maintained (i.e., by fully inserting all rods and ensuring the Rod Control System is incapable of rod withdrawal) yet do not initiate a feedwater isolation (P-4 and low T_{avg}) in MODE 3, consistent with Traveler TSTF-135.
- 3.3-123 This change deletes ACTION L.2 and renumbers L.3 since the requirement to close the unborated water source valves is not in the CTS and is not part of the current licensing basis. This new requirement is not applicable to DCPD which has a licensed dilution accident evaluation (refer to License Amendment 28/27). The current licensing bases in accordance with NUREG 0800, Section 15.4.6 provides adequate assurance that a dilution event will be recognized and arrested in a timely fashion.
- 3.3-124 Consistent with the current TS Table 4.3-1, Note [15], the note for ITS SR 3.3.1.4 is modified, a note is added to Table 3.3.1-1, and Function 20 are modified to clarify that the SR is required for the reactor trip bypass breaker local manual shunt trip only. The Bases for SR 3.3.1.14 clearly state that SR 3.3.1.14 includes the automatic undervoltage trip of the reactor trip bypass breakers. The Note (k) added to Table 3.3.1-1, Function 20 clarifies the Applicability of the undervoltage and shunt trip mechanisms to include those functions of the reactor trip bypass breakers when in use.

**CHANGE
NUMBER**

JUSTIFICATION

- 3.3-125 ITS SR 3.3.1.11 is modified by a Note that requires verification that the time constants are adjusted to the prescribed values. The addition of this Note is consistent with SR 3.3.1.10 and is required because SR 3.3.1.11 is used for the Power Range Neutron Flux - High Positive Rate [and High Negative Rate] trip functions which have a time constant associated with their calibration.
- 3.3-126 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-127 The MODE 2 applicability for the undervoltage RCP start of the steam-driven AFW pump is deleted and the surveillance Frequency is revised per the DCPP CTS. Thus, the Required Actions of ACTION I are revised to include entering MODE 2 for function 6.g and MODE 3 for function 5.b, and the required surveillance is changed from SR 3.3.2.7 to SR 3.3.2.8. This anticipatory start of the steam-driven AFW pump is not credited for MODE 2 operation, only the SG low level start signal is used for MODE 2 or 3.
- 3.3-128 This change revises ITS Table 3.3.4-1 to be consistent with CTS 3.3.3.5.
- 3.3-129 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-130 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-131 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-132 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-133 This change revises ITS LCO 3.3.5 and SR 3.3.5.3 to include the DG start sequence delay timers from CTS Table 3.3-4.
- 3.3-134 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.3-135 A MODE change restriction has been added to ITS 3.3.1 Condition C per the matrix discussed in CN 1-02-LS-1 of the 3.0 package (see LS-1 NSHC in the CTS Section 3/4.0, ITS Section 3.0 package).
- 3.3-136 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- {3.3-137 The Condition for Function 4.c is changed from Condition D to E consistent with the CTS. Plant design requires this Function to be bypassed, not tripped if inoperable.}

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(21 Pages)

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-01	This trip function or design feature is not included in the plant design or it is not credited and has no safety function.	Yes	Yes	Yes	Yes
3.3-02	It is acceptable to have more than one Turbine Stop Valve Closure - reactor trip function channels inoperable and placed in trip per CTS Table 3.3-1, Functional Unit [16.b], ACTION Statement [11]. In addition, the 4 hour bypass note applies only to the [Low Auto Stop Oil Pressure] channels. ITS 3.3.1 Condition P has been revised.	Yes	Yes	Yes	Yes
3.3-03	This change to ITS 3.3.1 Condition R is consistent with the current licensing basis. A 4-hour AOT for SSPS logic surveillance testing has little usefulness if the RTBs cannot be bypassed for the duration of that testing. RTB surveillance testing retains the current 2-hour AOT.	Yes	No, not in CTS.	No, not in CTS.	Yes (CTS per OL Amendment No. 64 dated 10-9-91)
3.3-04	This change represents the Callaway plant design as it relates to the SG Water Level-Low Low Environmental Allowance Modifier (EAM) and Trip Time Delay (TTD) circuitry. ITS Table 3.3.1-1 and Table 3.3.2-1 entries and the associated Required Actions have been enhanced to remove the redundancy in the CTS and add shutdown actions when inoperable channels aren't tripped per their Completion Time.	No, see CN 3.3-46.	No, not in CTS.	No, not in CTS.	Yes (CTS per OL Amendment No. 43 dated 4-14-89)
3.3-05	This change to ITS SR 3.3.1.3 Note 1 represents the CTS as it relates to the Overtemperature ΔT AFD f_1 (ΔI) penalty function.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes (CTS per OL Amendment No. 84 dated 11-8-93)
3.3-06	Retains CTS power requirement of 75% RTP in the ITS SR 3.3.1.6 Note concerning when the incore/excore calibration is performed. The ISTS proposal would require unnecessary delays in the post-refuel power ascension. The Note is further revised to permit achieving equilibrium conditions (per CTS 4.2.2.2.d.1) prior to performing the required surveillance.	Yes	No, see CN 3.3-97.	No, see 3.3-97.	No; see 3.3-97.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-07	Note 3 is added to ITS SR 3.3.1.11 to be consistent with the CTS Table 4.3-1 Note [5]. This ensures that this exception, for power and intermediate range detector plateau voltage verification as discussed in the ITS BASES for SR 3.3.1.11, is included in the Technical Specifications rather than being only found in the BASES.	Yes	Yes	Yes	Yes
3.3-08	Deletes the Reviewer's Note in ITS Tables 3.3.1-1 and 3.3.2-1 and adds a Note reflecting the Allowable Value as the LSSS. Trip Setpoints are listed in the Bases.	No, retained CTS format.	Yes	Yes	Yes
3.3-09	The addition of footnote [(m)] to ITS Table 3.3.1-1 for Function 10 clarifies the low flow setpoint relationship to the quantity identified as Minimum Measured Flow, consistent with the CTS.	Yes	No, not in CTS.	No, not in CTS.	Yes, (CTS per OL Amendment No. 15 dated 4-8-86)
3.3-10	[The Overtemperature ΔT setpoint equation had a bracket in the wrong place and was corrected.] In addition, the f_1 (ΔI) penalty function was corrected and the K_2 inequality sign was changed to an equal sign..	Yes	No, see CN 3.3-38.	Yes	Yes, (CTS per OL Amendment No. 102 dated 8-21-95)
3.3-11	Added "or Rod Control System incapable of rod withdrawal," which makes Note (f) the complete antithesis of Note (b).	Yes	No, see CN 3.3-41.	No, see CN 3.3-41.	No, see CN 3.3-41.
3.3-12	Corrects typo in the inequality sign of ITS Table 3.3.2-1 Note (h).	No, see CN 3.3-105.	Yes	Yes	Yes
3.3-13	The equations for Overtemperature ΔT and Overpower ΔT are revised to be consistent with the DCPD CTS. The value of the time constant τ_6 has always been 0 seconds and the factor utilizing the time constant has not been shown as part of the equation in licensing documents since the factor value has been unity. Thus, the factors utilizing the time constant has been deleted.	Yes	No	No	No
3.3-14	Retains the monthly COT for Function 6.h of ITS Table 3.3.2-1, per CTS Table 4.3-2 Functional Unit 6.h. No TADOT is performed.	No, not in CTS.	No, not in CTS.	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-15	The CHANNEL CHECK surveillance (SR 3.3.2.1) is deleted from the P-11 [] interlock because CHANNEL CHECKS are not applicable for permissive functions. This change is consistent with the CTS.	Yes	Yes	Yes	Yes
3.3-16	[Added Note (n) and] deleted SR 3.3.2.9 from Function 6.g in ITS Table 3.3.2-1. [Note (n) is added to avoid auxiliary feedwater actuations during normal plant startups and shutdowns.] These changes are consistent with CTS [Table 3.3-3 Note ### and] Table 4.3-2 Functional Unit [6.g].	No, not in CTS.	Yes	Yes, (Note [(p)] per CTS.)	Yes, (Note [(n)] per CTS, OL Amendment No. 26 dated 7-29-87)
3.3-17	New CONDITION [P] is added for Function 6.h of ITS Table 3.3.2-1, consistent with CTS Table 3.3-3 ACTION Statement [15].	No, not in CTS.	No, not in CTS.	Yes	Yes
3.3-18	Revises ITS 3.3.2 Condition K to be consistent with CTS Table 3.3-3 ACTION Statement 32, as revised per Enclosure 2, for Functional Unit 7.b.	No, not in CTS.	No, not in CTS.	No, see CN 3.3-134.	Yes, (CTS per OL Amendment No. 64 dated 10-9-91)
3.3-19	18-month test interval previously approved by NRC for selected slave relays which, if tested at power, could result in plant trips or upsets.	No, not in CTS.	No, not in CTS.	Yes (justified per SLNRC 84-0038 dated 2-27-84)	Yes (justified per SLNRC 84-0038 dated 2-27-84)
3.3-20	Added Note 2 on [Containment Radiation Level (High Range)] calibration in ITS SR [3.3.3.2] to be consistent with CTS Table [4.3-7 Note (2)].	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-21	ITS 3.3.3 was revised to reflect CTS 3.3.3.6. Containment Isolation Valve Position (and Notes) and Condensate Storage Tank Level were deleted. [Combined the power and source range neutron flux entries into a single neutron flux entry.] Consolidated the Thermocouple/Core Cooling Detection System entries. Changed number of required channels for RCS temperature to 2 for both hot and cold leg temperature. Changed Containment Pressure Wide Range to Containment Pressure Normal Range and added Refueling Water Storage Tank Level, Steam Line Pressure, and SG Water Level (Narrow Range) Functions as these are CTS requirements. Changed the required number of channels for SG Water Level (Wide Range) and AFW Flow Rate from 2 per SG to 1 per SG and added corresponding notes to Conditions A and C.	No, see CN 3.3-71.	No, see CN 3.3-70.	Yes (see Amendment No. 89)	Yes (see ULNRC-3023 dated 5-20-94 and Amendment No. 103 dated 10-20-95)
3.3-22	Consistent with CTS [3.3.3.5], [reactor trip breaker indication is excluded from CHANNEL CHECKS and from] CHANNEL CALIBRATIONS in ITS SR 3.3.4.3 since these SRs have no meaning for [this] function.	Yes	No, not in CTS.	Yes	Yes
3.3-23	CPSES-specific change that modifies ITS Surveillance Requirement 3.3.4.2 to include power circuits.	No	Yes	No	No
3.3-24	Changes ITS Table 3.3.4-1 to reflect CTS [3.3.3.5]. Deletes references to "controls" in the table (see also CN 3.3-94) and changes "Required Number of Functions" to "Required Channels" since the Table has been revised to include only instrumentation.	No, see CN 3.3-128.	Yes	Yes	Yes
3.3-25	Adds APPLICABILITY Note consistent with that found in ITS Table 3.3.1-1 (i.e., source range neutron flux is only required below the P-6 interlock). This is consistent with CTS Table 3.3-9.	No, see CN 3.3-84.	No, not in CTS.	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-26.	The Pressurizer PORV and Block Valve Controls are deleted and RCP Breaker Position and AFW Suction Pressure are added to ITS Table 3.3.4-1, consistent with the current licensing basis for compliance with GDC-19. The PORVs may be used for eventual plant cooldown; however, they are not required to attain HOT STANDBY which is the basis for the listed functions. The added functions may be used to ensure decay heat removal by the SGs in attaining HOT STANDBY.	No, not in CTS.	No, see CN 3.3-24.	Yes	Yes (per FSAR Section 7.4.3 and SER Sections 7.4.2 and 7.4.3.2)
3.3-27	This change modifies ITS SR 3.3.3.2 and SR 3.3.3.3 to allow for different surveillance frequencies for the hydrogen monitors than other PAMS components. The manufacturer for the CPSES hydrogen monitors specifies a more frequent calibration frequency than that required for the other PAMS instruments. The more frequent calibration is required to assure function operability.	No	Yes	No	No
3.3-28	Tie breaker changes per CTS Table 3.3-3, ACTION Statement 19 for Functional Unit 8.b.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes (CTS per OL Amendment No. 99 dated 4-18-95)
3.3-29	{Not used}	N/A	N/A	N/A	N/A
3.3-30	The portion of Condition C [(relabelled as D per CN 3.3-74)] referring to one or more functions with one or more automatic actuation trains inoperable is revised to cover BOP-ESFAS only.	No, not in CTS.	No, not in CTS.	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-31	The CTS require the response times associated with the [] containment [purge and exhaust] isolation function to be verified against the specific response time values contained in the [CTS]. The ITS is revised to match the CTS and the response time values are [moved to the FSAR]. As is done with the Reactor Trip System and the ESFAS instrumentation, this method is an appropriate way to control response times. [SR 3.3.6.8] is added to require the response time verifications.	Yes	Yes	No, will retain SRs in ITS 3.3.1 and 3.3.2 Tables.	Yes
3.3-32	ITS [3.3.6 ACTION A is modified by a Note and] Table 3.3.6-1 is changed to be consistent with CTS [3.3.2 Functional Unit 3.c and CTS 3.9.9]. Subfunctions [b, c and d] of Containment Radiation are stricken since only the gaseous [] channels provide the actuation function [and the bracketed setpoint is changed to reflect plant- specific requirements]. [The number of gaseous monitors required for CORE ALTERATIONS or during movement of irradiated fuel has been revised to one (either RM 44A or B) per the CTS]	Yes	No, see CN 3.3-73.	Yes	Yes
3.3-33	ITS Table 3.3.7-1 is revised to reflect the plant design. The CREVS is actuated by radiation monitors located in the air intakes, by a containment isolation - Phase A signal, by a containment purge isolation signal, by a fuel building ventilation isolation signal, or manually. The bracketed setpoint is revised to reflect CTS Table 3.3-6 requirements.	No, see CN 3.3-102.	No, see CN 3.3-78.	Yes	Yes
3.3-34	Revisions to add an LCO 3.0.3 exception Note 1 to ITS 3.3.8 reflect Industry Traveler TSTF-36, Rev. 2.	Yes	No, LCO does not apply.	Yes	Yes
3.3-35	ITS Table 3.3.8-1 is revised to reflect the plant design. Only the gaseous channels provide the actuation function. The bracketed setpoint is revised to reflect CTS Table 3.3-6 requirements.	No, not consistent with plant design.	No, LCO does not apply.	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-36	Revisions reflect revised BDMS setpoint in CTS.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes (CTS per OL Amendment No. 94 dated 3-7-95)
3.3-37	Several ITS Required Action Notes are modified to allow a channel to be placed in bypass for surveillance testing. [This change is consistent with the CTS.]	Yes	Yes	No, not in current design or TS.	No, not in current design or TS.
3.3-38	The CPSES design uses the N-16 based overtemperature and overpower protective functions. Several changes to the setpoints, Required Actions and Surveillances of NUREG-1431 are required to maintain the current licensing basis.	No	Yes	No	No
3.3-39	ITS Table 3.3.7-1 is changed to be consistent with CTS Table 3.3-3. The Actuation Logic was split to reflect the SSPS, with only MODE 1-4 Applicability, and BOP-ESFAS portions and associated SR requirements in the CTS.	No, not in CTS.	No, not in CTS.	Yes	Yes
3.3-40	Add "and setpoint adjustment" to ITS 3.3.1 Condition E, similar to the Note for Condition D.	Yes	Yes	Yes	Yes
3.3-41	ITS 3.3.1 Condition L is deleted to match the plant-specific design and the CTS for the Source Range Neutron Flux Function in MODES 3, 4, and 5 with the Rod Control System incapable of rod withdrawal and all rods fully inserted. Under these conditions, the source range instrumentation does not provide a Reactor Trip System Function. The source range channels provide only indication [and inadvertent boron dilution mitigation] when in this Applicability. Requirements related to the source range neutron flux channels in MODES 3, 4, and 5 when all rods are fully inserted and are not capable of being withdrawn have therefore been [moved to ITS 3.3.9. Footnote (f) of ITS Table 3.3-1 is added to Function 5 and revised accordingly].	No, see CN 3.3-123.	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-42	Delete ITS 3.3.1 Condition N and combine Functions 10.a and 10.b per Traveler TSTF-169.	Yes	Yes	Yes	Yes
3.3-43	Revise ITS 3.3.1 Condition R Notes 1 and 2 per Traveler TSTF-168. The 2-hour AOT should not be limited to only UVTA/STA maintenance.	Yes	Yes	Yes	Yes
3.3-44	Revise ITS 3.3.1 Conditions S and T and ITS 3.3.2 Condition L, as well as the number of Required Channels [or trains as Condition T applies to permissive P-7] in Tables 3.3.1-1 and 3.3.2-1, to reflect CTS ACTION Statements [8 and 20].	Yes	Yes	Yes	Yes
3.3-45	A new CONDITION and SR are added for the current licensing basis required seismic trip.	Yes	No, not in CTS.	No, not in CTS.	No, not in CTS.
3.3-46	A new CONDITION and SR are added for the current licensing basis required Steam Generator level low-low time delay trip. These changes affect both ITS 3.3.1 and 3.3.2.	Yes	No, not in CTS.	No, not in CTS.	No, not in CTS.
3.3-47	Note 2 of SR 3.3.1.2 is revised to limit the power increase to less than 30% per the current licensing basis before the SR is complete.	Yes	No, not in CTS.	No, not in CTS.	No, not in CTS.
3.3-48	ITS SR 3.3.1.7 has a NOTE that provides a four hour delay in the requirement to perform the Surveillance for source range instrumentation when entering MODE 3 from MODE 2. Wolf Creek has deleted this NOTE in accordance with current Surveillance Requirements and the revisions made in ITS Table 3.3-1. The requirements for this Surveillance will be maintained by SR 3.3.1.8 in Table 3.3.1-1 for each applicable Function. SR 3.3.1.8 has been structured to cover NI Functions specified in ITS Table 3.3.1-1 and SR 3.3.1.7 has been structured to cover all other Functions. This similar to how the NUREG has structured SR 3.3.1.10 and SR 3.3.1.11.	No	No	Yes	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-49	ITS SR 3.3.1.8 is revised to extend the conditional COT frequency for power and intermediate range channels from 4 hours after reducing power below P-10 to 12 hours, based on operating experience regarding the time needed to perform the COTs. It stands to reason that if 4 hours are allowed for 2 Source Range COTs, 12 hours should be allowed for 6 Intermediate Range and Power Range COTs.	Yes	Yes	Yes	Yes
3.3-50	ITS SR 3.3.1.12 is deleted per the CTS. Where cited in Table 3.3.2-1, a change to SR 3.3.1.10 has been made.	No, see CN 3.3-101.	Yes	Yes	Yes
3.3-51	ITS ACTION B.2 of LCO 3.3.7 is deleted, since DCPD cannot operate CRVS with both pressurization systems running at the same time.	Yes	No	No	No
3.3-52	Added Note [(I)] to ITS Table 3.3.1-1 per the CTS as an operator aid to note the dual RTS/ESFAS functions of SG Water Level Low-Low.	No, adopted ISTS format.	Yes	Yes	Yes
3.3-53	The REQUIRED CHANNELS description for Functions 2.a and 3.b.(1), of ITS Table 3.3.2-1, are revised per the DCPD CTS to note that only two switches (one per train) exist and that both must be moved coincident for manual initiation.	Yes	No	No	No
3.3-54	Function 18.b (P-7) of ITS Table 3.3.1-1 is clarified. COTs and Channel Calibrations apply to the P-10 and P-13 inputs, not to the P-7 logic function.	Yes	Yes	No, adopted ISTS format.	Yes
3.3-55	Revise ITS SR 3.3.1.16 and SR 3.3.2.10 to verify "required" response times, accommodating those channels that have no response time requirements per the current licensing basis. [As such, line-item references to these SRs in Tables 3.3.1-1 and 3.3.2-1 can be deleted. A similar revision to the ITS SR 3.3.6 SR note has also been made regarding SR 3.3.6.8.]	No, SRs will be retained in ITS Tables for required Functions.	Yes	Yes, SRs will be retained in ITS Tables for required Functions.	Yes
3.3-56	Revise ITS 3.3.2 Condition J to reflect CTS Table [3.3-3], ACTION Statement [19] for Functional Unit [6.g].	No, See CN 3.3-116.	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-57	Not used.	N/A	N/A	N/A	N/A
3.3-58	Adds new ITS 3.3.2 Condition [N] to reflect CTS Table 3.3-3 ACTION Statement [24] on manual AFW [and manual MSIV closure] initiation.	Yes	No, not in CTS.	No, adopted ISTS format.	Yes
3.3-59	Adds new ITS 3.3.2 Condition [R] to reflect CTS Table 3.3-3 ACTION Statement [21] on BOP-ESFAS portion of AFW initiation.	No, not in CTS.	No, not in CTS.	Yes	Yes
3.3-60	Consistent with the design and CTS, Surveillance Requirement[s] 3.3.2.3 and 3.3.2.7 are deleted as they are not used by any Function listed in Table 3.3.2-1.	Yes	Yes	No, used for BOP-ESFAS.	No, used for BOP-ESFAS.
3.3-61	Change ITS SR 3.3.2.11 Frequency to 18 months per CTS Table 4.3-2 Functional Unit [8.c], which is the ESFAS P-4 permissive. [Deleted the Note stating that verification of set point is not required per the CTS.]	Yes	Yes	Yes	Yes
3.3-62	Consistent with the CPSES design and CTS, isolation of the MSIVs also requires isolation of the associated upstream drip pot isolation valves.	No	Yes	No	No
3.3-63	Revise ITS Table 3.3.2-1 [Notes (b) and (g)] per CTS Table [3.3-3] Notes [# and ##]. This revision is a clarification to the operator that describes the circumstances under which the [Steamline Pressure Negative Rate - High, Steam Pressure-low, or Pressurizer Pressure-low functions may be or are] blocked [relative to the] P-11 permissive.	Yes	Yes	Yes	Yes
3.3-64	Revise ITS Table 3.3.2-1 Note (j) to exclude the MFRVs, consistent with CTS [3.7.1.6]. []	No, already in CTS.	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-65	A Note is added to the steam generator water level - high-high trip function to reflect the CPSES design and CTS. In the CPSES design, only three channels of the four steam generator water level signals provide input to this trip function. Therefore, in order to satisfy the single failure criterion, if one of these three channels is used as input to the Steam Generator Water Level Control System, its associated bistable must be placed in the tripped state.	No	Yes	No	No
3.3-66	The DCPD-specific MODE 4 requirement of the CTS is retained and added to Table 3.3.2-1 for SI actuated by Containment Pressure High.	Yes	No	No	No
3.3-67	In PAMS, add CPSES-specific operability requirements and Required Actions for the T-hot and T-cold indications consistent with both the current licensing basis and the intent of NUREG-1431. If a T-hot indication is unavailable, equivalent information is available from the Core Exit Temperature indication which is also a RG 1.97 variable. Similarly, if a T-cold indication is unavailable, equivalent information may be derived through the use of the steam generator pressure and steam tables, because the RCS cold leg temperature closely follows the steam generator saturation temperature.	No	Yes	No	No
3.3-68	A DCPD-specific Note is added to state that CONDITION D is only applicable in MODES 1 and 2. A new CONDITION H is added to require entering MODE 3 if CONDITION B is not met when entered due to not meeting CONDITION D.	Yes	No	No	No
3.3-69	{The phrase "...that is not normally energized" is deleted per the CTS. All of the instrumentation listed is normally energized at power.	Yes	No	No	No}
3.3-70	The PAM instrumentation list is modified to reflect the CPSES design and CTS.	No, see CN 3.3-71.	Yes	No, see CN 3.3-21.	No, see CN 3.3-21.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-71	This DCPP-specific change revises Table 3.3.3-1 per the Reviewers Note to update CTS PAM instruments per the requirements of Reg. Guide 1.97, and revises ACTIONS A and C to account for those functions with only one required channel.	Yes	No, see CN 3.3-70.	No, see CN 3.3-21.	No, see CN 3.3-21.
3.3-72	The Conditions and Required Actions related to two or more radiation monitors for the Containment Ventilation Isolation System are deleted to reflect the CPSES design and CTS. CTS ACTIONS for radiation monitors are retained in the ITS.	No	Yes	No	No
3.3-73	The CPSES CTS ACTIONS for inoperable Functions are retained in the ITS.	No	Yes	No	No
3.3-74	Revise ITS 3.3.6 Conditions A, B, and C to reflect current licensing basis and add a new Condition applicable during Core Alterations or movement of irradiated fuel assemblies within containment to allow a 4 hour Completion Time to restore one inoperable radiation monitoring channel prior to isolating the purge pathway. This new Condition represents a relaxation in a previously relocated Action Statement; however, that Action Statement inappropriately calls for the same measures whether one or both purge radiation monitors are inoperable.	No, not in CTS.	No, not in CTS.	No, retained CTS.	Yes (per FSAR Section 16.3.3.7B)
3.3-75	The DCPP-specific CHANNEL FUNCTIONAL TEST is substituted for the COT per current licensing basis.	Yes	No	No	No
3.3-76	Consistent with the current design and TS, a Trip Actuating Device Operational Test (TADOT) is not required for any of the functions explicitly listed in Table 3.3.6-1; therefore, the associated Surveillance Requirement is deleted. Note that a TADOT is required in accordance with LCO 3.3.2 for functions 3.a.1 and 2.a, as referenced in the Table.	Yes	Yes	No, used for Manual Initiation Function 1.	No, used for Manual Initiation Function 1.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-77	Containment Vent Isolation is initiated by the ESFAS Phase "A" isolation signals. As such, the number of required channels and required surveillances for the manual initiation of Containment Vent Isolation are captured by the requirements for Phase "A" isolation in the ESFAS tables.	Yes	Yes	No, not in CTS.	No, not in CTS.
3.3-78	The modifications to the ITS are consistent with both the intent of the ITS and the CPSES CREFS design and CTS.	No	Yes	No	No
3.3-79	Add APPLICABILITY columns to ITS Tables 3.3.6-1 and 3.3.7-1 to reflect CTS with varying Functional Applicabilities.	Yes	Yes	Yes	Yes
3.3-80	Revise Condition B and add new Condition C to both ITS 3.3.7 and ITS 3.3.8 to reflect current ACTIONS [30 and 38 of Table 3.3-6].	No, not in CTS.	No, see CN 3.3-78.	No-see CN 3.3-81.	Yes
3.3-81	This change revises condition B and adds Condition C to both ITS 3.3.7 and ITS 3.3.8 to reflect CTS ACTIONS 27 and 30 of Table 3.3-6.	No	No	Yes	No
3.3-82	The Conditions, Required Actions, etc. are revised per the DCPD current licensing basis. The plant FBACS does not perform any accident mitigation functions except during the fuel handling accident.	Yes	No	No	No
3.3-83	Add Note [(c)] to ITS Table 3.3.8-1 to reflect the CTS.	No, see CN 3.3-82.	No, LCO does not apply.	Yes	Yes
3.3-84	The Note of SR 3.3.4.4 is deleted since Table 3.3.4-1 does not have a neutron detector specified per the DCPD CTS.	Yes	No	No	No
3.3-85	The Note in Required Action 3.3.5 A.1 would be deleted as it is inconsistent with the current CPSES design and TS.	No	Yes	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-86	Surveillance Requirement 3.3.5.2 is revised to reflect the current CPSES plant design and licensing basis. A Note is added to SR 3.3.5.2 indicating that setpoint verification is not applicable for the performance of the TADOT. This verification is performed during Channel Calibrations (see SR 3.3.5.3).	No	Yes, see also CNs 3.3-31, 3.3-130, and 3.3-131.	No	No
3.3-87	Not used.	NA	NA	NA	NA
3.3-88	Revise ITS 3.3.9 to apply in MODE 2 only below P-6 and to reflect ACTION Statement 5.b per CTS Table 3.3-1.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
3.3-89	Revise COT in ITS SR 3.3.9.3 to add the 4 hour allowance from ITS SR 3.3.1.7.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
3.3-90	Exclude neutron detectors from CHANNEL CALIBRATION ITS SR 3.3.9.4 per CTS Table 4.3-1, Functional Unit 6, Note 4.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
3.3-91	Add CHANNEL CHECK and response time surveillances (ITS SR 3.3.9.1 and SR 3.3.9.5) per CTS Table 4.3-1, Functional Unit 6, Note 12.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
3.3-92	Adds SR 3.3.4.2 Note that the ASP controls for the TDAFW pump and SG ASDs are not required to be verified prior to entry into MODE 3, consistent with CTS SR 4.3.3.5.3.	No, adopted ISTS format.	No, not in CTS.	No, adopted ISTS format.	Yes
3.3-93	ITS 3.3.1 Condition V is deleted. It is not entered from Table 3.3.1-1 nor do the Bases clarify when it would be needed, raising the concern of misinterpretation. Condition V does not replace LCO 3.0.3 requirements to assess when the plant is outside the licensing basis.	Yes	Yes	Yes	Yes
3.3-94	ITS 3.3.4 is revised per CTS [3.3.3.5] with regard to [ASP] controls.	Yes	Yes	No, adopted ISTS format.	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-95	ITS 3.3.1 Condition H, Required ACTION H.1, and the second part of Function 4 Applicability (MODE 2 below P-6) in ITS 3.3.1 are deleted since they provide no real compensatory measures. [With their deletion, there is no need to repeat the > P-6 Applicability in Conditions F and G.] In accordance with LCO 3.0.4, the intermediate range detectors must be OPERABLE prior to entering the Applicability of the retained part of Function 4 (i.e., MODE 2 above P-6). Condition H and Required ACTION H.1 ensure the same thing and, therefore, can be deleted. This change is consistent with Traveler TSTF-135.	Yes	Yes	Yes	Yes
3.3-96	[] Note 2 for ITS SR 3.3.1.3 is revised to replace the bracketed 15% RTP power level constraint with 50% RTP. The specified power level in ITS SR 3.3.1.3 should reflect the applicable safety analysis basis consistent with the [APPLICABILITY and] Required Actions of ITS LCO 3.2.3 (AFD) and LCO 3.2.4 (QPTR).	Yes	Yes	Yes	Yes
3.3-97	The NOTE for SR 3.3.1.6 of ITS 3.3.1 has been revised to state "Not required to be performed until 72 hours after achieving equilibrium conditions with THERMAL POWER \geq 75% RTP". This is consistent with the CTS Functional Unit 2.a surveillance requirement of Table 4.3-1 as modified by Note (6), and current operating experience of 72 hours for performing the SR.	No, adopted bracketed ISTS time of 24 hours; see also CN 3.3-06.	Yes	Yes	Yes
3.3-98	This change reflects the Callaway-specific modification of the main steam and feedwater isolation system (MSFIS).	No	No	No	Yes, per OL Amendment No. 117 dated 10-1-96.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-99	<p>ITS 3.3.5 Condition B is revised to allow 12 hours for one bus to restore the instrument function to the capability of continued operation in Condition A. The 12 hour allowance is based on the ITS 3.8.1.F allowance for an inoperable sequencer.</p> <p>[A new Condition C is added to ITS 3.3.5 to cover multiple channel inoperability on both ESF buses. The Completion Time of 1 hour is provided to restore the functional capability of Condition B.]</p> <p>A new Condition [D] is also added to the ACTIONS of ITS LCO 3.3.5. The new Condition provides the appropriate shutdown actions if Conditions [A, B, or C] are not satisfied when in MODES 1-4. Condition [D] is consistent with the ACTIONS required by the AC Sources ITS 3.8.1 for Modes 1-4.</p> <p>Condition C of NUREG-1431 Rev. 1 is revised to be Condition [E]. Condition [E] provides the appropriate default Condition for failure to satisfy Conditions [A, B, or C] when the loss of power instrumentation is required operable to support the diesel generator required operable in ITS LCO 3.8.2.</p>	No, see CN 3.3-104.	No, see CN 3.3-131.	Yes	Yes
3.3-100	Not used.	N/A	N/A	N/A	N/A
3.3-101	The Note for ITS SR 3.3.1.12 is deleted since the plant design no longer includes the RTD bypass. The SR is retained and is applied to the required seismic trip instrumentation per the current licensing basis. Where cited in Table 3.3.2-1, a change to SR 3.3.1.10 has been made.	Yes	No, see CN 3.3-50.	No, see CN 3.3-50.	No, see CN 3.3-50.
3.3-102	The control room (CR) does not have CR Atmosphere monitors as part of its current design. There are redundant CR intake monitors for each intake.	Yes	No, see CN 3.3-78.	No, see CN 3.3-33.	No, see CN 3.3-33.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-103	Function 11 of ITS Table 3.3.1-1 is revised per the DCPD CTS to reflect the current plant design of only a two loop trip. With this revision Condition O is no longer used, since it was only applicable to the single loop trip.	Yes	No	No	No
3.3-104	CONDITION A of ITS 3.3.5 is revised to incorporate CTS ACTIONS 15 and 16. CONDITIONS B and C are not used.	Yes	No, see CN 3.3-131.	No, see CN 3.3-99.	No, see CN 3.3-99.
3.3-105	Function 4.d.(2) of ITS Table 3.3.2-1 and notes (c) and (h) are revised per the DCPD CTS.	Yes	No, see CN 3.3-12.	No, see CN 3.3-12.	No, see CN 3.3-12.
3.3-106	Delete ISTS Required Actions B.2.2 and U.2.2. These Required Actions are not needed due to exiting the APPLICABILITY via Required Actions B.2.1 and U.2.1.	Yes	Yes	Yes	Yes
3.3-107	Based upon operating experience to change Thermal Power in a controlled fassion without challenging the plant and consistent with the CTS which does not have a Completion Time for restoring one channel to OPERABLE sttus; but does pervent going above P-10 until it is restored, the Completion Time for ITS 3.3.1 Required Actions F.1 and F.2 should be increased to 24 hours.	Yes	Yes	Yes	Yes
3.3-108	Not used.	N/A	N/A	N/A	N/A
3.3-109	Not used.	N/A	N/A	N/A	N/A
3.3-110	Not used.	N/A	N/A	N/A	N/A
3.3-111	Add a Note to ITS SR 3.3.1.7 for source range instrumentation to verify interlocks P-6 and P-10 are in their required state for existing unit conditions. This is consistent with the CTS.	Yes	Yes	No-see CN 3.3-48.	Yes
3.3-112	Not used.	N/A	N/A	N/A	N/A
3.3-113	Not used.	N/A	N/A	N/A	N/A
3.3-114	Not used.	N/A	N/A	N/A	N/A

JLS Conversion to Improved Technical Specifications Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File .

CTS 3/4.4 - Reactor Coolant System

ITS 3.4 - Reactor Coolant System



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.4

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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

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CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.1.1	LCO		01-01-LG	3.4.4	LCO		
3.4.1.1	APP			3.4.4	APP		
3.4.1.1	ACTION			3.4.4	CONDITION	A	
4.4.1.1	SR		01-01-LG	3.4.4.1	SR		
3.4.1.2	LCO		01-03-LS	3.4.5	LCO	a	
3.4.1.2	LCO		01-03-LS	3.4.5	LCO	b	
3.4.1.2	LCO	Note 1	01-06-M	3.4.5	LCO	Note a	
3.4.1.2	LCO	Note 2	01-06-M	3.4.5	LCO	Note b	
3.4.1.2	LCO	a	01-01-LG	3.4.5	BASES		
3.4.1.2	LCO	b		3.4.5	BASES		
3.4.1.2	LCO	c		3.4.5	BASES		
3.4.1.2	LCO	d		3.4.5	BASES		
3.4.1.2	APP			3.4.5	APP		
3.4.1.2	ACTION	a		3.4.5	CONDITION	A	
3.4.1.2	ACTION	a		3.4.5	CONDITION	B	
3.4.1.2	ACTION	b	01-03-LS 03-14-LS	3.4.5	CONDITION	C.2	
3.4.1.2	ACTION	c	01-03-LS	3.4.5	CONDITION	D.2 & D.3	3.4-31
3.4.1.2	ACTION	c	01-03-LS	3.4.5	CONDITION	D.1	3.4-31
3.4.1.2	ACTION	c	01-03-LS	3.4.5	CONDITION	D.2	
3.4.1.2	ACTION	c	01-03-LS	3.4.5	CONDITION	D.3	
4.4.1.2.1	SR			3.4.5.3	SR		
4.4.1.2.2	SR			3.4.5.2	SR		
4.4.1.2.3	SR		01-01-LG	3.4.5.1	SR		
3.4.1.3	LCO			3.4.6	LCO		
3.4.1.3	LCO	Note	01-06-M	3.4.6	LCO	Note 1a	3.4-01
3.4.1.3	LCO	Note	01-06-M	3.4.6	LCO	Note 1b	
3.4.1.3	LCO	a	01-01-LG	3.4.6	BASES		
3.4.1.3	LCO	b		3.4.6	BASES		
3.4.1.3	LCO	c		3.4.6	BASES		
3.4.1.3	LCO	d		3.4.6	BASES		
3.4.1.3	LCO	e		3.4.6	BASES		
3.4.1.3	LCO	f		3.4.6	BASES		
3.4.1.3	LCO	Note	01-17-LG	3.4.6	LCO	Note 2	3.4-10
3.4.1.3	LCO	Note	01-17-LG	3.4.6	LCO	Note 2	3.4-19
3.4.1.3	APP			3.4.6	APP		
3.4.1.3	ACTION	a		3.4.6	CONDITION	A	
3.4.1.3	ACTION	b	01-07-M	3.4.6	CONDITION	C.1	

CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.1.3	ACTION	b	01-07-M	3.4.6	CONDITION	C.2	
4.4.1.3.1	SR			3.4.6.3	SR		
4.4.1.3.2	SR			3.4.6.2	SR		
4.4.1.3.3	SR		01-01-LG	3.4.6.1	SR		
New	LCO	Note **	01-08-LS	3.4.7	LCO	Note 4	
3.4.1.4.1	LCO	Note *	01-06-M	3.4.7	LCO	Note 1	
3.4.1.4.1	LCO	Note *1	01-06-M	3.4.7	LCO	Note 1a	
3.4.1.4.1	LCO	Note *2	01-06-M	3.4.7	LCO	Note 1b	
3.4.1.4.1	LCO	a		3.4.7	LCO	a	
3.4.1.4.1	LCO	Note #		3.4.7	LCO	Note 2	
3.4.1.4.1	LCO	b		3.4.7	LCO	b	
3.4.1.4.1	APP			3.4.7	APP		
3.4.1.4.1	APP	Note ##	01-17-LG	3.4.7	LCO	Note 3	3.4-10
3.4.1.4.1	APP	Note ##	01-17-LG	3.4.7	LCO	Note 3	3.4-19
3.4.1.4.1	ACTION	a		3.4.7	CONDITION	A.1	
3.4.1.4.1	ACTION	a		3.4.7	CONDITION	A.2	
3.4.1.4.1	ACTION	b	01-10-M	3.4.7	CONDITION	B.1	
3.4.1.4.1	ACTION	b	01-10-M	3.4.7	CONDITION	B.2	
4.4.1.4.1.1	SR			3.4.7.2	SR		
4.4.1.4.1.2	SR		01-01-LG	3.4.7.1	SR		
New	SR		01-11-M	3.4.7.3	SR		
3.4.1.4.2	LCO			3.4.8	LCO		
3.4.1.4.2	LCO	Note #		3.4.8	LCO	Note 2	
3.4.1.4.2	LCO	Note *	01-06-M	3.4.8	LCO	Note 1	3.4-03 3.4-01
3.4.1.4.2	LCO	Note 1		3.4.8	LCO	Note 1b	
3.4.1.4.2	LCO	Note 2		3.4.8	LCO	Note 1a	3.4-04
3.4.1.4.2	LCO	Note 3		3.4.8	LCO	Note 1c	
3.4.1.4.2	APP			3.4.8	APP		
3.4.1.4.2	ACTION	a	01-12-M	3.4.8	CONDITION	A	
3.4.1.4.2	ACTION	b	01-10-M	3.4.8	CONDITION	B.1	
3.4.1.4.2	ACTION	b	01-10-M	3.4.8	CONDITION	B.2	
4.4.1.4.2	SR		01-01-LG	3.4.8.1	SR		
New	SR		01-11-M 01-16-M	3.4.8.2	SR		
3.4.2.2	LCO			3.4.10	LCO		
3.4.2.2	LCO	Note *	02-04-LG	3.4.10	BASES		
3.4.2.2	APP		01-17-LG	3.4.10	APP		3.4-10

CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.2.2	APP	Note #	01-17-LG	3.4.10	APP		3.4-10
3.4.2.2	ACTION	a	01-17-LG	3.4.10	CONDITION	A	
3.4.2.2	ACTION	b		3.4.10	APP	Note	
3.4.2.2	ACTION	b		3.4.10	CONDITION	B.1	
New	ACTION	c	02-03-M	3.4.10	CONDITION	B.2	
New	ACTION	c	01-17-LG	3.4.10	CONDITION	B.2	
4.4.2.2	SR			3.4.10.1	SR		
3.4.3	LCO			3.4.9	LCO	a	
3.4.3	LCO		03-05-M	3.4.9	LCO	b	
3.4.3	APP			3.4.9	APP		
3.4.3	ACTION	a		3.4.9	CONDITION	B	
3.4.3	ACTION	a		3.4.9	CONDITION	C.1	
3.4.3	ACTION	a		3.4.9	CONDITION	C.2	
3.4.3	ACTION	b	03-04-LS	3.4.9	CONDITION	A.1	
3.4.3	ACTION	b	03-04-LS	3.4.9	CONDITION	A.2	3.4-31
3.4.3	ACTION	b	03-04-LS	3.4.9	CONDITION	A.3	3.4-31
3.4.3	ACTION	b	03-04-LS	3.4.9	CONDITION	A.4	
4.4.3.1	SR			3.4.9.1	SR		
4.4.3.2	SR		03-02-LG	3.4.9.2	SR		
4.4.3.2	SR		03-03-LS	3.4.9.2	SR		3.4-17
4.4.3.3	SR			3.4.9.3	SR		
3.4.4	LCO			3.4.11	LCO		
3.4.4	APP			3.4.11	APP		
3.4.4	APP	Note	04-01-LS	3.4.11	ACTION	Note 1	
3.4.4	ACTION	a	04-02-LS	3.4.11	CONDITION	A	
3.4.4	ACTION	a	04-05-LS	3.4.11	CONDITION	D.1	
3.4.4	ACTION	a	04-05-LS	3.4.11	CONDITION	D.2 & D.3	
3.4.4	ACTION	b	04-02-LS	3.4.11	CONDITION	B.1	
3.4.4	ACTION	b	04-02-LS	3.4.11	CONDITION	B.2	
3.4.4	ACTION	b	04-02-LS	3.4.11	CONDITION	E.2	3.4-39
3.4.4	ACTION	b1	04-05-LS	3.4.11	CONDITION	B.3	
3.4.4	ACTION	b2	04-02-LS	3.4.11	CONDITION	E.1	3.4-39
3.4.4	ACTION	b2	04-03-M	3.4.11	CONDITION	E.1	3.4-21
3.4.4	ACTION	b2	04-05-LS	3.4.11	CONDITION	E.4 & E.5	3.4-39
3.4.4	ACTION	c1	04-06-LS	3.4.11	CONDITION	C.1	
3.4.4	ACTION	c1	04-06-LS	3.4.11	CONDITION	C.2	3.4-21
3.4.4	ACTION	c2	04-07-LS	3.4.11	CONDITION	F.1	

CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.4	ACTION	c2	04-07-LS	3.4.11	CONDITION	F.2	3.4-21
3.4.4	ACTION	c2	04-07-LS	3.4.11	CONDITION	F.3	3.4-21
3.4.4	ACTION	c3	04-03-M	3.4.11	CONDITION	G.1	3.4-39
3.4.4	ACTION	c3	04-05-LS	3.4.11	CONDITION	G.2	
3.4.4	ACTION	d		3.4.11	ACTION	Note 2	
4.4.4.1	SR	a		3.4.11.2	SR		
4.4.4.1	SR	b	04-04-LG 04-08-LS			Not Used	
4.4.4.2	SR		04-08-LS 04-09-LS	3.4.11.1	SR		3.4-47 3.4-35
4.4.4.3	SR	a		3.4.11.3	SR	a	3.4-26
4.4.4.3	SR	b		3.4.11.3	SR	b	3.4-26
4.4.4.3	SR	c		3.4.11.3	SR	c	3.4-26
3.4.5	LCO		05-01-A	5.5.9			
3.4.5	LCO		05-02-A	5.5.9			
3.4.5	APP		05-01-A	5.5.9			
3.4.5	ACTION		05-01-A	5.5.9			
4.4.5.0	SR		05-01-A	5.5.9			
4.4.5.0	SR		05-01-A	5.5.9			
4.4.5.1	SR		05-01-A 05-02-A	5.5.9			
4.4.5.2	SR	a	05-01-A	5.5.9			
4.4.5.2	SR	b1	05-01-A	5.5.9			
4.4.5.2	SR	b2	05-01-A	5.5.9			
4.4.5.2	SR	b3	05-01-A	5.5.9			
4.4.5.2	SR	c1	05-01-A	5.5.9			
4.4.5.2	SR	c2	05-01-A	5.5.9			
4.4.5.3	SR	a	05-01-A	5.5.9			
4.4.5.3	SR	b	05-01-A	5.5.9			
4.4.5.3	SR	c1	05-01-A	5.5.9			
4.4.5.3	SR	c2	05-01-A	5.5.9			
4.4.5.3	SR	c3	05-01-A	5.5.9			
4.4.5.3	SR	c4	05-01-A	5.5.9			
4.4.5.4	SR	a1	05-01-A	5.5.9			
4.4.5.4	SR	a2	05-01-A	5.5.9			
4.4.5.4	SR	a3	05-01-A	5.5.9			
4.4.5.4	SR	a4	05-01-A	5.5.9			
4.4.5.4	SR	a5	05-01-A	5.5.9			
4.4.5.4	SR	a6	05-01-A	5.5.9			
4.4.5.4	SR	a7	05-01-A	5.5.9			

CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
4.4.5.4	SR	a8	05-03-A				
4.4.5.4	SR	a9	05-01-A	5.5.9			
4.4.5.4	SR	b	05-01-A 05-02-A	5.5.9			
4.4.5.5	SR	a	05-01-A	5.5.9			
4.4.5.5	SR	b1	05-01-A	5.5.9			
4.4.5.5	SR	b2	05-01-A	5.5.9			
4.4.5.5	SR	b3	05-01-A	5.5.9			
4.4.5.5	SR	c	05-01-A	5.5.9			
Table 4.4-1			05-01-A	5.5.9			
Table 4.4-2			05-01-A	5.5.9			
3.4.6.1	LCO	a		3.4.15	LCO	B	3.4-14
3.4.6.1	LCO	b		3.4.15	LCO	A	
3.4.6.1	LCO	c		3.4.15	LCO	C	
3.4.6.1	APP			3.4.15	APP		3.4-15
3.4.6.1	ACTION	Note(new)	06-23-LS	3.4.15	ACTION	Note	
3.4.6.1	ACTION	Note(new)	06-23-LS	3.4.15	ACTION	Note	
3.4.6.1	ACTION	Note(new)	06-23-LS	3.4.15	ACTION	Note	3.4-15
3.4.6.1	ACTION	Note(new)	06-23-LS	3.4.15	ACTION	Note	3.4-14
3.4.6.1	ACTION			3.4.15	CONDITION	A.2	
3.4.6.1	ACTION	(new)	06-01-M	3.4.15	CONDITION	A.1	
3.4.6.1	ACTION	(new)	06-02-LS	3.4.15	CONDITION	B.1.1	
3.4.6.1	ACTION			3.4.15	CONDITION	B.1.2	
3.4.6.1	ACTION			3.4.15	CONDITION	B.2	
3.4.6.1	ACTION			3.4.15	CONDITION	C.1.1	3.4-16
3.4.6.1	ACTION			3.4.15	CONDITION	C.1.2	3.4-16
3.4.6.1	ACTION			3.4.15	CONDITION	C.2.1	3.4-14
3.4.6.1	ACTION			3.4.15	CONDITION	C.2.2	
3.4.6.1	ACTION	Note *	06-26-LS	3.4.15	CONDITION	A Note	3.4-36
3.4.6.1	ACTION	(new)	06-26-LS	3.4.15	CONDITION	B.1.2 Note	3.4-36
3.4.6.1	ACTION	Note *	06-26-LS	3.4.15	CONDITION	C.1.1 Note	3.4-36
4.4.6.1	SR	a	06-03-A	3.4.15.1	SR		3.4-14
4.4.6.1	SR	a		3.4.15.2	SR		3.4-29
4.4.6.1	SR	a		3.4.15.2	SR		3.4-14
4.4.6.1	SR	a		3.4.15.4	SR		
4.4.6.1	SR	b		3.4.15.3	SR		
4.4.6.1	SR	c		3.4.15.5	SR		

CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.6.2	LCO	a		3.4.13	LCO	a	
3.4.6.2	LCO	b		3.4.13	LCO	b	
3.4.6.2	LCO	c		3.4.13	LCO	d	
3.4.6.2	LCO	c		3.4.13	LCO	e	
3.4.6.2	LCO	d		3.4.13	LCO	c	
3.4.6.2	LCO	e	06-06-A	3.5.5	LCO		
3.4.6.2	LCO	f	06-07-LG				
3.4.6.2	LCO	f	06-25-LS	3.4.14.1	SR		
3.4.6.2	APP			3.4.13	APP		
3.4.6.2	APP (new)	Note #1	06-08-LS	3.4.14	APP		
3.4.6.2	APP (new)	Note #2	06-09-LS				
3.4.6.2	ACTION	a		3.4.13	CONDITION	B.1	
3.4.6.2	ACTION	a		3.4.13	CONDITION	B.2	
3.4.6.2	ACTION	b	06-09-LS	3.4.13	CONDITION	A	
3.4.6.2	ACTION	c	06-11-LS	3.4.14	CONDITION	A.1	
3.4.6.2	ACTION	c	06-11-LS	3.4.14	CONDITION	A.2.1	
3.4.6.2	ACTION	(new)	06-12-M	3.4.14	CONDITION	A.2.2	
3.4.6.2	ACTION	c	06-12-M	3.4.14	CONDITION	B.1	
3.4.6.2	ACTION	c	06-12-M	3.4.14	CONDITION	B.2	
3.4.6.2	ACTION	c##(new)	06-12-M	3.4.14	CONDITION	A Note	
4.4.6.2.1	SR	a	06-13-LS		Not used		
4.4.6.2.1	SR	b	06-13-LS		Not used		
4.4.6.2.1	SR	c	06-14-A	3.5.5.1	SR		
4.4.6.2.1	SR	d	06-17-LG	3.4.13.1	SR		
4.4.6.2.1	SR	d	06-26-LS	3.4.13.1	SR Note		
4.4.6.2.1	SR	e	06-16-LS		Not used		
4.4.6.2.2	SR		06-07-LG				
4.4.6.2.2	SR		06-20-A	5.5.9			
4.4.6.2.2	SR	a		3.4.14.1	SR		
4.4.6.2.2	SR	b	06-19-TR3				
4.4.6.2.2	SR	c		3.4.14.1	SR		
Table 3.4-1			06-07-LG				
3.4.8	LCO	a		3.4.16	LCO		
3.4.8	LCO	b		3.4.16	LCO		
3.4.8	APP		08-01-LS	3.4.16	APP		
3.4.8	APP	Note *	08-01-LS	3.4.16	APP		
3.4.8	ACTION	Note(new)	08-02-LS	3.4.16	CONDITION	A Note	
3.4.8	ACTION	a		3.4.16	CONDITION	A.1	
3.4.8	ACTION	a		3.4.16	CONDITION	A.2	
3.4.8	ACTION	a		3.4.16	CONDITION	C	3.4-39

CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.8	ACTION	b		3.4.16	CONDITION	B	
3.4.8	ACTION		08-01-LS	3.4.16	APP		
4.4.8	SR			3.4.16.1	SR		
4.4.8	SR			3.4.16.2	SR		
Figure 3.4-1				Figure 3.4.16-1			
Table 4.4-4	ITEM	1	08-03-LS	3.4.16.1	SR		
Table 4.4-4	ITEM	1	08-01-LS	3.4.16	APP		
Table 4.4-4	ITEM	1 Note	08-08-LG	3.4.16	BASES		
Table 4.4-4	ITEM	2		3.4.16.2	SR		
Table 4.4-4	ITEM	3		3.4.16.3	SR		
Table 4.4-4	ITEM	3 ***	08-06-A	1.1	DEFINITIONS		
Table 4.4-4	ITEM	3 *	08-07-A	3.4.16.3	SR	Note	
Table 4.4-4	ITEM	4a	08-04-M	3.4.16	CONDITION	A.1	
Table 4.4-4	ITEM	4a	08-04-M	3.4.16	CONDITION	A.2	
Table 4.4-4	ITEM	4a	08-01-LS	3.4.16	APP		
Table 4.4-4	ITEM	4b	08-05-LS	3.4.16.2	SR		
Table 4.4-4	Note #		08-06-A	3.4.16	CONDITION	C	
3.4.9.1	LCO		09-01-LG	3.4.3	LCO		
3.4.9.1	LCO	a	09-01-LG		(PTLR)		
3.4.9.1	LCO	b	09-01-LG		(PTLR)		
3.4.9.1	LCO	c	09-01-LG		(PTLR)		
3.4.9.1	APP			3.4.3	APP (5.6.6)		
3.4.9.1	ACTION		09-02-M	3.4.3	CONDITION	A.1& B.1	
3.4.9.1	ACTION		09-03-M	3.4.3	CONDITION	A.2& B.2	
3.4.9.1	ACTION		09-02-M	3.4.3	CONDITION	C.1	
3.4.9.1	ACTION	Note *	09-02-M	3.4.3	CONDITION	C.2	
4.4.9.1	SR		09-16-M	3.4.3.1	SR		
Figure 3.4-2			09-01-LG		(PTLR)		
Figure 3.4-3			09-01-LG		(PTLR)		
3.4.9.3	LCO		09-06-M	3.4.12	LCO		
3.4.9.3	LCO	a	09-01-LG	3.4.12	LCO	a	3.4-21
3.4.9.3	LCO	b		3.4.12	LCO	b	
3.4.9.3	APP		01-17-LG	3.4.12	APP		3.4-10
3.4.9.3	ACTION	a		3.4.12	CONDITION	E	3.4-21
3.4.9.3	ACTION	a		3.4.12	CONDITION	G	3.4-21

CROSS-REFERENCE TABLE FOR 3/4.4
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Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.9.3	ACTION	b		3.4.12	CONDITION	F	3.4-21
3.4.9.3	ACTION	b		3.4.12	CONDITION	F	3.4-27
3.4.9.3	ACTION	b		3.4.12	CONDITION	G	3.4-21
3.4.9.3	ACTION	c		3.4.12	CONDITION	G	3.4-21
3.4.9.3	ACTION	d	09-07-TR			Not used	
3.4.9.3	ACTION	New	09-09-M	3.4.12	CONDITION	B	3.4-06
3.4.9.3	ACTION	Note *	09-08-LS	3.4.12	CONDITION	B	3.4-06
3.4.9.3	ACTION	New	09-10-M	3.4.12	APP	Note 1	
3.4.9.3	ACTION	New	09-10-M	3.4.12	CONDITION	C	
3.4.9.3	ACTION	New	09-10-M	3.4.12	CONDITION	D.2	
3.4.9.3	ACTION	New	01-17-LG	3.4.12	CONDITION	D.1	3.4-10
3.4.9.3	ACTION	New	09-11-M	3.4.12	CONDITION	G	3.4-21
3.4.9.3	Action	New	09-15-M	3.4.12	CONDITION	A	
4.4.9.3.1	SR	a	09-12-LS	3.4.12.8	SR		3.4-10 3.4-49
4.4.9.3.1	SR	b		3.4.12.9	SR		
4.4.9.3.1	SR	c		3.4.12.6	SR		
4.4.9.3.2	SR			3.4.12.5	SR		3.4-49
New	SR		09-14-M	3.4.12.3	SR		3.4-23
4.5.3.2	SR		09-15-M	3.4.12.1	SR		
4.5.3.2	SR		09-09-M	3.4.12.2	SR		

CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.2.5	LCO			3.4.1	LCO		
3.2.5	LCO			3.4.1	LCO	a	
3.2.5	LCO			3.4.1	LCO	b	
3.2.5	LCO			3.4.1	LCO	c	3.4-41
				3.4.1	APP		
				3.4.1	APP	Note	
3.2.5	ACTION			3.4.1	CONDITION	A	
3.2.5	ACTION			3.4.1	CONDITION	B	
4.2.5.1	SR			3.4.1.1	SR		
4.2.5.1	SR			3.4.1.2	SR		
4.2.5.1	SR			3.4.1.3	SR		3.4-41
4.2.5.4	SR			3.4.1.4	SR		3.4-51
				3.4.1.4	SR		3.4-38
				Table 3.4.1-1			3.4-41
				Table 3.4.1-2			3.4-41
3.1.1.4	LCO			3.4.2	LCO		
				3.4.2	APP		
3.1.1.4	ACTION			3.4.2	CONDITION	A	3.4-32
4.1.1.4	SR			3.4.2.1	SR		3.4-33
3.4.9.1	LCO		09-01-LG	3.4.3	LCO		
3.4.9.1	APP			3.4.3	APP		
3.4.9.1	ACTION		09-02-M	3.4.3	CONDITION	A.1	
3.4.9.1	ACTION		09-03-M	3.4.3	CONDITION	A.2	
				3.4.3	CONDITION	B.1	
				3.4.3	CONDITION	B.2	
3.4.9.1	ACTION		09-02-M	3.4.3	CONDITION	C.1	
3.4.9.1	ACTION	Note *	09-02-M	3.4.3	CONDITION	C.2	
4.4.9.1	SR		09-16-M	3.4.3.1	SR		
3.4.1.1	LCO		01-01-LG	3.4.4	LCO		
3.4.1.1	APP			3.4.4	APP		
3.4.1.1	ACTION			3.4.4	CONDITION	A	
4.4.1.1	SR		01-01-LG	3.4.4.1	SR		
3.4.1.2	LCO		01-03-LS	3.4.5	LCO	a	3.4-01
3.4.1.2	LCO		01-03-LS	3.4.5	LCO	b	
3.4.1.2	LCO	Note 1	01-06-LS	3.4.5	LCO	Note a	

CROSS-REFERENCE TABLE FOR 3/4.4
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Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.1.2	LCO	Note 2	01-06-M	3.4.5	LCO	Note b	
3.4.1.2	APP			3.4.5	APP		
3.4.1.2	ACTION	a		3.4.5	CONDITION	A	
3.4.1.2	ACTION	a		3.4.5	CONDITION	B	
3.4.1.2	ACTION	b	01-03-LS 03-14-LS	3.4.5	CONDITION	C.1	
3.4.1.2	ACTION	c	01-03-LS	3.4.5	CONDITION	C.2	3.4-31
3.4.1.2	ACTION	c	01-03-LS	3.4.5	CONDITION	D.1	3.4-31
3.4.1.2	ACTION	c	01-03-LS	3.4.5	CONDITION	D.2	
3.4.1.2	ACTION	c	01-03-LS	3.4.5	CONDITION	D.3	
4.4.1.2.3	SR		01-01-LG	3.4.5.1	SR		
4.4.1.2.2	SR			3.4.5.2	SR		
4.4.1.2.1	SR			3.4.5.3	SR		
3.4.1.3	LCO			3.4.6	LCO		
3.4.1.3	LCO	Note	01-06-M	3.4.6	LCO	Note 1a	3.4-01
3.4.1.3	LCO	Note	01-06-M	3.4.6	LCO	Note 1b	
3.4.1.3	LCO	Note	01-17-LG	3.4.6	LCO	Note 2	3.4-10
3.4.1.3	LCO	Note	01-17-LG	3.4.6	LCO	Note 2	3.4-19
3.4.1.3	APP			3.4.6	APP		
3.4.1.3	ACTION	a		3.4.6	CONDITION	A	
3.4.1.3	ACTION	b		3.4.6	CONDITION	B Note	3.4-02
3.4.1.3	ACTION	a		3.4.6	CONDITION	B	
3.4.1.3	ACTION	b	01-07-M	3.4.6	CONDITION	C.1	
3.4.1.3	ACTION	b	01-07-M	3.4.6	CONDITION	C	
4.4.1.3.3	SR		01-01-LG	3.4.6.1	SR		
4.4.1.3.2	SR			3.4.6.2	SR		
4.4.1.3.1	SR			3.4.6.3	SR		
3.4.1.4.1	LCO	a		3.4.7	LCO	a	
3.4.1.4.1	LCO	b		3.4.7	LCO	b	
3.4.1.4.1	LCO	Note *	01-06-M	3.4.7	LCO	Note 1	3.4-01
3.4.1.4.1	LCO	Note *1	01-06-M	3.4.7	LCO	Note 1a	
3.4.1.4.1	LCO	Note *2	01-06-M	3.4.7	LCO	Note 1b	
3.4.1.4.1	LCO	Note #		3.4.7	LCO	Note 2	
3.4.1.4.1	APP	Note ##	01-17-LG	3.4.7	LCO	Note 3	3.4-10
3.4.1.4.1	APP	Note ##	01-17-LG	3.4.7	LCO	Note 3	3.4-19
New	LCO	Note **	01-08-LS	3.4.7	LCO	Note 4	
3.4.1.4.1	APP			3.4.7	APP		
3.4.1.4.1	ACTION	a		3.4.7	CONDITION	A.1	
3.4.1.4.1	ACTION	a		3.4.7	CONDITION	A.2	

CROSS-REFERENCE TABLE FOR 3/4.4
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Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.1.4.1	ACTION	b	01-10-M	3.4.7	CONDITION	B.1	
3.4.1.4.1	ACTION	b	01-10-M	3.4.7	CONDITION	B.2	
4.4.1.4.1.2	SR		01-01-LG	3.4.7.1	SR		
4.4.1.4.1.1	SR			3.4.7.2	SR		
New	SR		01-11-M 01-16-M	3.4.7.3	SR		
3.4.1.4.2	LCO			3.4.8	LCO		
3.4.1.4.2	LCO	Note *	01-06-M	3.4.8	LCO	Note 1	3.4-03 3.4-01
3.4.1.4.2	LCO	Note 2		3.4.8	LCO	Note 1a	3.4-04
3.4.1.4.2	LCO	Note 1		3.4.8	LCO	Note 1b	
3.4.1.4.2	LCO	Note 3		3.4.8	LCO	Note 1c	
3.4.1.4.2	LCO	Note #		3.4.8	LCO	Note 2	
3.4.1.4.2	APP			3.4.8	APP		
3.4.1.4.2	ACTION	a	01-12-M	3.4.8	CONDITION	A	
3.4.1.4.2	ACTION	b	01-10-M	3.4.8	CONDITION	B.1	
3.4.1.4.2	ACTION	b	01-10-M	3.4.8	CONDITION	B.2	
4.4.1.4.2	SR		01-01-LG	3.4.8.1	SR		
New	SR		01-11-M	3.4.8.2	SR		
3.4.3	LCO			3.4.9	LCO	a	
3.4.3	LCO		03-05-M	3.4.9	LCO	b	
3.4.3	APP			3.4.9	APP		
3.4.3	ACTION	b	03-04-LS	3.4.9	CONDITION	A.1	
3.4.3	ACTION	b	03-04-LS	3.4.9	CONDITION	A.2	3.4-31
3.4.3	ACTION	b	03-04-LS	3.4.9	CONDITION	A.3	3.4-31
3.4.3	ACTION	b	03-04-LS	3.4.9	CONDITION	A.4	
3.4.3	ACTION	a		3.4.9	CONDITION	B	
3.4.3	ACTION	a		3.4.9	CONDITION	C.1	
3.4.3	ACTION	a		3.4.9	CONDITION	C.2	
4.4.3.1	SR			3.4.9.1	SR		
4.4.3.2	SR		03-02-LG	3.4.9.2	SR		
4.4.3.2	SR		03-03-LS	3.4.9.2	SR		3.4-17
4.4.3.3	SR			3.4.9.3	SR		
3.4.2.2	LCO			3.4.10	LCO		
3.4.2.2	APP		01-17-LG	3.4.10	APP		3.4-10
3.4.2.2	APP	Note #	01-17-LG	3.4.10	APP		3.4-10
3.4.2.2	ACTION	b		3.4.10	APP	Note	
3.4.2.2	ACTION	a	01-17-LG	3.4.10	CONDITION	A	

CROSS-REFERENCE TABLE FOR 3/4.4
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Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.2.2	ACTION	b		3.4.10	CONDITION	B.1	
New	ACTION	c	02-03-M	3.4.10	CONDITION	B.2	
New	ACTION	c	01-17-LG	3.4.10	CONDITION	B.2	3.4-10
4.4.2.2	SR			3.4.10.1	SR		
3.4.4	LCO			3.4.11	LCO		
3.4.4	APP			3.4.11	APP		
3.4.4	APP	Note	04-01-LS	3.4.11	ACTION	Note 1	
3.4.4	ACTION	d		3.4.11	ACTION	Note 2	
3.4.4	ACTION	a	04-02-LS	3.4.11	CONDITION	A	
3.4.4	ACTION	b	04-02-LS	3.4.11	CONDITION	B.1	
3.4.4	ACTION	b	04-02-LS	3.4.11	CONDITION	B.2	
3.4.4	ACTION	b1	04-05-LS	3.4.11	CONDITION	B.3	3.4-21
3.4.4	ACTION	c1	04-06-LS	3.4.11	CONDITION	C.1	
3.4.4	ACTION	c1	04-06-LS	3.4.11	CONDITION	C.2	3.4-21
3.4.4	ACTION	a	04-05-LS	3.4.11	CONDITION	D.1	3.4-21
3.4.4	ACTION	a	04-05-LS	3.4.11	CONDITION	D.2	3.4-39
3.4.4	ACTION	b2	04-02-LS	3.4.11	CONDITION	E.1	3.4-39
3.4.4	ACTION	b2	04-03-M	3.4.11	CONDITION	E.1	3.4-21
3.4.4	ACTION	b	04-02-LS	3.4.11	CONDITION	E.2	3.4-39
3.4.4	ACTION	b2	04-05-LS	3.4.11	CONDITION	E.3	3.4-39
3.4.4	ACTION	b2	04-05-LS	3.4.11	CONDITION	E.4	3.4-39
3.4.4	ACTION	c2	04-07-LS	3.4.11	CONDITION	F.1	
3.4.4	ACTION	c2	04-07-LS	3.4.11	CONDITION	F.2	3.4-21
3.4.4	ACTION	c2	04-07-LS	3.4.11	CONDITION	F.3	3.4-21
3.4.4	ACTION	c3	04-03-M	3.4.11	CONDITION	G.1	3.4-39
3.4.4	ACTION	c3	04-05-LS	3.4.11	CONDITION	G.2	
				3.4.11.1	SR	Note	3.4-47 3.4-35
4.4.4.2	SR		04-08-LS 04-09-LS	3.4.11.1	SR		
4.4.4.1	SR	a	04-08-LS	3.4.11.2	SR		
4.4.4.3	SR	a		3.4.11.3	SR	a	3.4-26
4.4.4.3	SR	b		3.4.11.3	SR	b	3.4-26
4.4.4.3	SR	c		3.4.11.3	SR	c	3.4-26
		NA		3.4.11.3	SR		
		NA		3.4.11.4	SR		
3.4.9.3	LCO		09-06-M	3.4.12	LCO		
3.5.3.1	LCO	a	03-02-LS	3.4.12	LCO		
3.5.3.2	LCO		04-01-LS	3.4.12	LCO		3.4-45

CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.9.3	LCO	a	09-01-LG	3.4.12	LCO	a1	3.4-21
3.4.9.3	LCO	a		3.4.12	LCO	a2	
3.4.9.3	LCO	a		3.4.12	LCO	a3	
3.4.9.3	LCO	b		3.4.12	LCO	b	
3.4.9.3	ACTION	New	01-17-LG	3.4.12	APP		
3.4.9.3	APP		01-17-LG	3.4.12	APP		3.4-10
3.4.9.3	ACTION	New	09-10-M	3.4.12	APP	Note 1	
				3.4.12	APP	Note 2	3.4-01
3.4.9.3	ACTION	New	09-09-M	3.4.12	CONDITION	A	3.4-06
3.5.3.2	ACTION		04-01-LS	3.4.12	CONDITION	A	
				3.4.12	CONDITION	B Note	3.4-06
3.4.9.3	ACTION	Note *	09-08-LS	3.4.12	CONDITION	B	
3.4.9.3	ACTION	New	09-10-M	3.4.12	CONDITION	C	
3.4.9.3	ACTION	New	01-17-LG	3.4.12	CONDITION	D.1	3.4-10
3.4.9.3	ACTION	New	09-10-M	3.4.12	CONDITION	D.2	
3.4.9.3	ACTION	a		3.4.12	CONDITION	E	3.4-21
3.4.9.3	ACTION	b		3.4.12	CONDITION	F	3.4-21
3.4.9.3	ACTION	b		3.4.12	CONDITION	F	3.4-27
3.4.9.3	ACTION	a		3.4.12	CONDITION	G	3.4-21
3.4.9.3	ACTION	b		3.4.12	CONDITION	G	3.4-21
3.4.9.3	ACTION	c		3.4.12	CONDITION	G	3.4-21
3.4.9.3	ACTION	New	09-11-M	3.4.12	CONDITION	G	3.4-21
3.5.3.2	ACTION		04-02-M	3.4.12	CONDITION	G	
New	SR		09-15-M	3.4.12.1	SR		
4.5.3.2	SR		04-03-M	3.4.12.1	SR		
New	SR		09-09-M	3.4.12.2	SR		
4.1.2.3.3	SR			3.4.12.2	SR		
4.5.3.1.2	SR		03-08-LG	3.4.12.2	SR		
4.5.3.1.2	SR		03-09-M	3.4.12.2	SR		
New	SR		09-14-M	3.4.12.3	SR		3.4-23
				3.4.12.4	SR		
4.4.9.3.2	SR			3.4.12.5	SR	Note	3.4-49
4.4.9.3.2	SR			3.4.12.5	SR		
4.4.9.3.2	SR			3.4.12.5	SR		3.4-28
4.4.9.3.1	SR	c		3.4.12.6	SR		3.4-21
		NA		3.4.12.7	SR		
				3.4.12.8	SR	Note	3.4-10 3.4-49
4.4.9.3.1	SR	a	09-12-LS	3.4.12.8	SR		3.4-21
4.4.9.3.1	SR	b		3.4.12.9	SR		3.4-21

CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.6.2	LCO	a		3.4.13	LCO	a	
3.4.6.2	LCO	b		3.4.13	LCO	b	
3.4.6.2	LCO	d		3.4.13	LCO	c	
3.4.6.2	LCO	c		3.4.13	LCO	d	
3.4.6.2	LCO	c		3.4.13	LCO	e	
3.4.6.2	APP			3.4.13	APP		
3.4.6.2	ACTION	b	06-09-LS	3.4.13	CONDITION	A	
3.4.6.2	ACTION	a		3.4.13	CONDITION	B.1	
3.4.6.2	ACTION	a		3.4.13	CONDITION	B.2	
4.4.6.2.1	SR	d	06-26-LS	3.4.13.1	SR Note		3.4-36
4.4.6.2.1	SR	d	06-17-LG	3.4.13.1	SR		
4.0.6	SR		01-14-A	3.4.13.2	SR		
				3.4.14	LCO		
3.4.6.2	APP	Note #1	06-08-LS	3.4.14	APP		
				3.4.14	ACTION	Note 1	
				3.4.14	ACTION	Note 2	
3.4.6.2	ACTION	c ##	06-12-M	3.4.14	CONDITION	A Note	
3.4.6.2	ACTION	c	06-11-LS	3.4.14	CONDITION	A.1	
3.4.6.2	ACTION	c	06-11-LS	3.4.14	CONDITION	A.2.1	
3.4.6.2	ACTION	c	06-12-M	3.4.14	CONDITION	A.2.2	
3.4.6.2	ACTION	c	06-12-M	3.4.14	CONDITION	B.1	
3.4.6.2	ACTION	c	06-12-M	3.4.14	CONDITION	B.2	
3.5.2	ACTION		N/A	3.4.14	CONDITION	C.1	N/A
				3.4.14.1	SR	Note 1	
				3.4.14.1	SR	Note 2	
				3.4.14.1	SR	Note 3	
4.4.6.2.2	SR	a		3.4.14.1	SR		3.4-24
4.4.6.2.2	SR	c		3.4.14.1	SR		3.4-42
				3.4.14.2	SR	Note	
4.5.2	SR		N/A	3.4.14.2	SR		N/A
				3.4.14.3	SR	Note	
				3.4.14.3	SR		
3.3.3.1	LCO	(LD)		3.4.15	LCO	a	
3.4.6.1	LCO	b		3.4.15	LCO	a	
3.3.3.1	LCO	(LD)		3.4.15	LCO	b	
3.4.6.1	LCO	a		3.4.15	LCO	b	3.4-14
3.4.6.1	LCO	c		3.4.15	LCO	c	
3.4.6.1	APP			3.4.15	APP		3.4-15
3.4.6.1	ACTION	Note	06-23-LS	3.4.15	CONDITION	A Note	

CROSS-REFERENCE TABLE FOR 3/4.4
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
3.4.6.1	ACTION	Note *	06-26-LS	3.4.15	CONDITION	A Note	3.4-36
3.3.3.1	ACTION	a (LD)		3.4.15	CONDITION	A	
3.4.6.1	ACTION		06-01-M	3.4.15	CONDITION	A.1	
3.4.6.1	ACTION			3.4.15	CONDITION	A.2	
3.3.3.1	ACTION	b (LD)		3.4.15	CONDITION	B	
3.3.3.1	ACTION	c (LD)		3.4.15	CONDITION	B	
Table 3.3-4	FUNCTIONAL	1.a		3.4.15	CONDITION	B	
Table 3.3-4	ACTION	29		3.4.15	CONDITION	B	
3.4.6.1	ACTION	Note	06-23-LS	3.4.15	CONDITION	B Note	
3.4.6.1	ACTION		06-02-LS	3.4.15	CONDITION	B.1:1	3.4-14
3.4.6.1	ACTION	NOTE	06-26-LS	3.4.15	CONDITION	B.1.2 Note	3.4-36
3.4.6.1	ACTION			3.4.15	CONDITION	B.1.2	
3.4.6.1	ACTION			3.4.15	CONDITION	B.2	3.4-14
				3.4.15	CONDITION	B.2.2	
				3.4.15	CONDITION	C.1	3.4-36 3.4-16
				3.4.15	CONDITION	C.2	3.4-14
3.3.3.1	ACTION	c (LD)		3.4.15	CONDITION	D	
Table 3.3-4	FUNCTIONAL	1.b		3.4.15	CONDITION	D	
3.4.6.1	ACTION	Note	06-23-LS	3.4.15	CONDITION	D Note	3.4-15
3.4.6.1	ACTION	Note	06-23-LS	3.4.15	CONDITION	D Note	
3.4.6.1	ACTION			3.4.15	CONDITION	D.1.1	3.4-16
3.4.6.1	ACTION	Note *	06-26-LS	3.4.15	CONDITION	D.1.1 Note	3.4-36
3.4.6.1	ACTION			3.4.15	CONDITION	D.1.2	3.4-16
3.4.6.1	ACTION			3.4.15	CONDITION	D.1	3.4-14
3.4.6.1	ACTION			3.4.15	CONDITION	D.2.2	
				3.4.15	CONDITION	E.1	
				3.4.15	CONDITION	E.2	
				3.4.15	CONDITION	F	
4.3.3.1	SR	a (LD)		3.4.15.1	SR		
4.4.6.1	SR	a	06-03-A	3.4.15.1	SR		3.4-14
4.4.6.2.1	SR	a	06-13-LS	3.4.15.1	SR		
4.4.6.2.1	SR	b	06-13-LS	3.4.15.1	SR		
4.3.3.1	SR	c (LD)		3.4.15.2	SR		
4.4.6.1	SR	a		3.4.15.2	SR		3.4-29
4.4.6.1	SR	a		3.4.15.2	SR		3.4-14
4.4.6.1	SR	b		3.4.15.3	SR		
4.4.6.1	SR	a		3.4.15.4	SR		

CROSS-REFERENCE TABLE FOR 3.4
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para	Change	Item	Code	Para	Change
4.4.6.1	SR	c		3.4.15.5	SR		
3.4.8	LCO	a		3.4.16	LCO		
3.4.8	LCO	b		3.4.16	LCO		
3.4.8	APP		08-01-LS	3.4.16	APP		
3.4.8	APP	Note *	08-01-LS	3.4.16	APP		
3.4.8	ACTION		08-01-LS	3.4.16	APP		
Table 4.4-4	ITEM	1	08-01-LS	3.4.16	APP		
Table 4.4-4	ITEM	4a	08-01-LS	3.4.16	APP		
3.4.8	ACTION	Note	08-02-LS	3.4.16	CONDITION	A Note	
3.4.8	ACTION	a		3.4.16	CONDITION	A.1	
Table 4.4-4	ITEM	4a	08-04-M	3.4.16	CONDITION	A.1	
3.4.8	ACTION	.a		3.4.16	CONDITION	A.2	
Table 4.4-4	ITEM	4a	08-04-M	3.4.16	CONDITION	A.2	
				3.4.16	CONDITION	B.1	3.4-22
3.4.8	ACTION	b		3.4.16	CONDITION	B.2	3.4-25
3.4.8	ACTION	a		3.4.16	CONDITION	C	
Table 4.4-4	Note #		08-06-A	3.4.16	CONDITION	C	
4.4.8	SR			3.4.16.1	SR		
Table 4.4-1	ITEM	1	08-03-LS	3.4.16.1	SR		
				3.4.16.2	SR	Note	
4.4.8	SR			3.4.16.2	SR		
Table 4.4-4	ITEM	2		3.4.16.2	SR		
Table 4.4-1	ITEM	4b	08-05-LS	3.4.16.2	SR		
Table 4.4-4	ITEM	3 *	08-07-A	3.4.16.3	SR	Note	
Table 4.4-4	ITEM	3		3.4.16.3	SR		
Figure 3.4-1				Figure 3.4.16-1			
		NA		3.4.17		NA	
		NA		3.4.18		NA	
		NA		3.4.19		NA	

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

Methodology for Cross-Reference Tables
(Continued)

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).

Note: The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
3.4.1.1	3/4 4-1
3.4.1.2	3/4 4-2
3.4.1.3	3/4 4-3
3.4.1.4.1	3/4 4-5
3.4.1.4.2	3/4 4-6
3.4.2.1	N/A
3.4.2.2	3/4 4-8
3.4.3	3/4 4-9
3.4.4	3/4 4-10
3.4.5	3/4 4-11
3.4.6.1	3/4 4-18
3.4.6.2	3/4 4-19
3.4.7	N/A
3.4.8	3/4 4-25
3.4.9.1	3/4 4-30
3.4.9.2	N/A
3.4.9.3	3/4 4-35
Methodology	(2 Pages)

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be OPERABLE and in operation.

01:01:LG

APPLICABILITY: MODES 1 and 2.

ACTION:

With less than the above required reactor coolant loops OPERABLE and in operation, be in at least HOT STANDBY within 6 hours.

01:01:LG

SURVEILLANCE REQUIREMENT

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours.

01:01:LG

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with and either two reactor coolant loops in operation when the rod control system is capable of rod withdrawal. Reactor Trip System breakers are closed and or one reactor coolant loop in operation when the rod control system is not capable of rod withdrawal. * Reactor Trip System breakers are open.*

01-03-LS1

- a. Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump.
- b. Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump.
- c. Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump, and
- d. Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump.

01-01-LG

APPLICABILITY: MODE 3.

ACTION:

- a. With one less than the above required reactor coolant loops inoperable OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.
- b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, and rod control system capable of rod withdrawal, within 1 hour restore two loops to operation OR place the rod control system in a condition incapable of rod withdrawal, open the Reactor Trip System breakers.
- c. With no RCS loops OPERABLE OR no reactor coolant loop in operation, immediately place the rod control system in a condition incapable of rod withdrawal and suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

ED

03-04-LS29

01-03-LS1

01-03-LS1

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (continued)

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 15% at least once per 12 hours.

4.4.1.2.3 The required reactor coolant loop(s) shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

01:01:LG

*All reactor coolant pumps may be deenergized ~~removed from operation~~ for up to 1 hour ~~per 8 hour period~~ provided:

01:16:A

01:06:M

(1) no operations are permitted which could cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops/trains listed below shall be OPERABLE and at least one of these loops/trains shall be in operation:*

- a. ~~Reactor Coolant Loop 1 and its associated steam generator and reactor coolant pump, **~~ 01-01:LG
- b. ~~Reactor Coolant Loop 2 and its associated steam generator and reactor coolant pump, **~~
- c. ~~Reactor Coolant Loop 3 and its associated steam generator and reactor coolant pump, **~~
- d. ~~Reactor Coolant Loop 4 and its associated steam generator and reactor coolant pump, **~~
- e. Residual Heat Removal (RHR) Train 1, and
- f. Residual Heat Removal (RHR) Train 2.

APPLICABILITY: MODE 4.

ACTION:

- a. ~~With one less than the above required loops/trains inoperable OPERABLE, immediately initiate corrective action to return the required loop/train to OPERABLE status as soon as possible; if the remaining OPERABLE loop/train is an RHR train, be in COLD SHUTDOWN within 24 hours.~~ ED
- b. ~~With no loops OPERABLE or no loop/train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return one the required coolant loop to OPERABLE status and operation /train to operation.~~ 01-07:M

~~*All reactor coolant pumps and RHR pumps may be deenergized removed from operation for up to 1 hour per 8 hour period provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.~~ 01-16:A
01-08:M

~~**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 270°F the temperature below which LTOP is required as specified in the PTLR unless: (1) the pressurizer water level is less than 50%, or (2) secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.~~ 01-17-LG

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s) and/or RHR pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 15% at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant loop or RHR train shall be verified to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours.

0101EG

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) train shall be OPERABLE and in operation*. **and either:

01-08-LS2

- a. One additional RHR train shall be OPERABLE#, or
- b. The secondary side water level of at least two steam generators shall be greater than or equal to 15%.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

- a. With one of the RHR trains inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR train to OPERABLE status or restore the required steam generator water level as soon as possible.
- b. With ~~required RHR trains inoperable or~~ no RHR train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR train to OPERABLE status and operation to operation.

01-10-M

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

01-01-EG

~~(NEW) Verify correct breaker alignment and indicated power are available to the RHR pump that is not in operation at least once per 7 days~~

01-11-M

* The RHR pump may be deenergized ~~removed from operation~~ for up to ~~1~~ 1 hour per 8 hour period provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

01-16-A

01-06-M

One RHR train may be inoperable for up to 2 hours for surveillance testing provided the other RHR train is OPERABLE and in operation.

A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 270°F ~~the temperature below which~~

SURVEILLANCE REQUIREMENTS (continued)

~~LTOP is required as specified in the PTLR unless:~~

~~01:17:EG~~

(1) the pressurizer water level is less than 50%, or (2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

~~** All RHR Loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.~~

~~01:08:LS2~~

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) trains shall be OPERABLE# and at least one RHR train shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

- a. With ~~one~~ less than the above required RHR trains ~~inoperable~~ OPERABLE, immediately initiate corrective action to return the required trains to OPERABLE status. ~~as soon as possible.~~ 01:12:M

- b. With ~~required RHR trains inoperable~~ OR no RHR train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR train ~~to OPERABLE status and operation~~ to operation. 01:10:M

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one RHR train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours. 01:01:LG

~~(new) Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation at least once per 7 days.~~ 01:11:M

One RHR train may be inoperable for up to 2 hours for surveillance testing provided the other RHR train is OPERABLE and in operation.

* The RHR pump may be ~~de-energized~~ removed from operation for up to ~~1~~ 1 hour per ~~8~~ hour period provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, (2) core outlet temperature is maintained at least 10°F below saturation temperature, and (3) ~~no draining operations to further reduce the RCS water volume are permitted.~~ the reactor vessel water level is above the vessel flange. 01:16:A
01:08:M
01:20:LS27

~~This Page Intentionally Deleted~~

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig \pm 1%.*

APPLICABILITY: MODES 1, 2, and 3 and 4-#

~~MODE 4 with all RCS cold leg temperatures > the temperature below which LTOP is required as specified in the PTLR~~

01:17:LG

ACTION:

- a. With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN with any all RCS cold leg temperatures less than or equal to 270°F the temperature below which LTOP is required as specified in the PTLR within the following 6 hours.
- b. The provisions of Specification 3.0.4 may be suspended for up to 18 hours per valve for entry into and during operations in MODE 3 and 4-# for the purpose of setting the pressurizer Code safety valves under ambient (hot) conditions provided a preliminary cold setting was made prior to heatup.

01:17:LG

~~c. With two or more pressurizer safety valves inoperable, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN with any RCS cold leg temperature less than or equal to the temperature below which LTOP is required as specified in the PTLR within the following 6 hours.~~

02:03:M

01:17:LG

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

~~* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.~~

02:04:LG

~~# When all RCS cold leg temperatures are greater than 270°F.~~

01:17:LG

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1600 cubic feet (90% of span) and two groups of pressurizer heaters each having a capacity of at least 150 kW and capable of being powered from an emergency power supply.

03-05-M

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With the pressurizer water level not within limit otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip Breakers open all rods fully inserted and the Rod Control System incapable of rod withdrawal within 6 hours and in HOT SHUTDOWN within the following 6 hours.

03-01-LS4

03-04-LS29

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by measuring heater group power at least once per 92 days. 18 months.

03-02-LG

03-03-LS28

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the heaters.

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

NOTE: Separate condition entry is allowed for each valve.

04-01-LS5

ACTION:

a. With one or more PORV(s) inoperable because of excessive seat leakage and capable of being manually cycled, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and maintain power to the block valve(s), otherwise, continue restoration activities and be in at least HOT STANDBY within the next 6 hours and reduce Tavg to < 500°F and HOT SHUTDOWN within the following 6 hours.

04-02-LS6

04-05-LS31

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, and not capable of being manually cycled, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s), and

04-02-LS6

1. With only one Class 1 PORV OPERABLE inoperable, and not capable of being manually cycled, restore at least a total of two Class 1 PORVs to OPERABLE status within the following 72 hours or continue restoration activities and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN and reduce Tavg to < 500°F within the following 6 hours, or

04-05-LS31

2. With no both Class 1 PORVs OPERABLE inoperable and not capable of being manually cycled, immediately initiate action to restore at least one Class 1 PORV to OPERABLE status within 1 hour and follow ACTION b.1, above, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable Class 1 PORV or continue restoration activities and be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN and reduce Tavg to < 500°F within the following 6 hours.

04-02-LS6

04-03-M

04-05-LS31

c. 1. With one or more block valve(s) inoperable, within 1 hour, place associated PORV in manual control

04-06-LS32

AND

1. Restore the block valve(s) to OPERABLE status within 72 hours, or

2. With more than one block valve inoperable, place associated PORVs in manual control within 1 hour

04-07-LS33

AND

Restore one block valve for a Class 1 PORV to OPERABLE status within 2 hours

REACTOR COOLANT SYSTEM

ACTION: (continued)

AND

restore the remaining block valves for a Class 1 PORV to OPERABLE status within 72 hours. (If the remaining block valve is for a non-Class 1 PORV, close the block valve and remove the power.)

OR

~~2. Close the PORV(s) and remove power from its associated solenoid.~~

~~Also, comply with ACTION b, as appropriate, for the isolated PORV(s).~~

~~3. If required action and associated completion times of c.1. or c.2. are not met, continue restoration activities and be in HOT STANDBY within the next 6 hours AND be in HOT SHUTDOWN and reduce Tavg to < 500°F within the following 6 hours.~~

04:05:LS31

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the PORV through one complete cycle of full travel* during MODES 3 or 4 with the block valves closed, and

04:08:LS34

~~b. Performing a CHANNEL CALIBRATION of the actuation instrumentation.~~

04:04:LG

4.4.4.2 In addition to the requirements of Specification 4.0.5, each block valve shall be demonstrated OPERABLE at least once per 92 days* by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION a, b, or c. in Specification 3.4.4.

04:08:LS34

04:09:LS36

4.4.4.3 The safety-related nitrogen supply for the PORVs shall be demonstrated OPERABLE at least once per 18 months by:

a. Isolating and venting the normal air supply, and

b. Verifying that any leakage of the Class 1 Backup Nitrogen System is within its limits, and

c. Operating the PORVs through one complete cycle of full travel.

* Only required to be performed in MODES 1 and 2.

04:08:LS34

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

05:01A

LIMITING CONDITION FOR OPERATION

~~3.4.5 Each steam generator shall be OPERABLE.~~

05:02A

~~APPLICABILITY: MODES 1, 2, 3 and 4.~~

ACTION:

~~With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.~~

SURVEILLANCE REQUIREMENTS

~~4.4.5.0 Each Steam generator tube integrity steam generator shall be demonstrated OPERABLE by performance of the following augmented Inservice inspection program, and the requirement of Specification 4.0.5.~~

05:01A

~~4.4.5.1 Steam Generator Sample Selection and Inspection Each Steam generator tube integrity shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4.1.~~

05:01A

05:02A

~~4.4.5.2 Steam Generator Tube Sample Selection and Inspection The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9.2. The Inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.9.c. 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9d. 4.4.5.4. The tubes selected for each Inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:~~

05:01A

- ~~a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;~~
- ~~b. The first sample of tubes selected for each Inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:
 - ~~1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%);~~
 - ~~2) Tubes in those areas where experience has indicated potential problems, and~~
 - ~~3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.~~~~

SURVEILLANCE REQUIREMENTS (continued)

~~c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:~~

- ~~1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and~~
- ~~2) The inspections include those portions of the tubes where imperfections were previously found.~~

~~The results of each sample inspection shall be classified into one of the following three categories:~~

<u>Category</u>	<u>Inspection Results</u>
G-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
G-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
G-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.
<u>Note:</u>	In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (continued)

~~4.4.5.3 Inspection Frequencies~~ The above required Inservice inspections of steam generator tubes shall be performed at the following frequencies:

- a. ~~The first Inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent Inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;~~
- b. ~~If the results of the Inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a. The interval may then be extended to a maximum of once per 40 months; and~~
- e. ~~Additional, unscheduled Inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:~~
 - 1) ~~Reactor to secondary tube leaks (not including leaks originating from tube to tube sheet welds) in excess of the limits of Specification 3.4.6.2; or~~
 - 2) ~~A seismic occurrence greater than the Double Design Earthquake, or~~
 - 3) ~~A loss of coolant accident requiring actuation of the Engineered Safety Features, or~~
 - 4) ~~A main steam line or feedwater line break.~~

SURVEILLANCE REQUIREMENTS (continued)4.4.5.4 Acceptance Criteria

a. As used in this Specification:

- 1) ~~Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;~~
- 2) ~~Degradation means a service induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;~~
- 3) ~~Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;~~
- 4) ~~% Degradation means the percentage of the tube wall thickness affected or removed by degradation;~~
- 5) ~~Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;~~
- 6) ~~Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;~~
- 7) ~~Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of a Double Design Earthquake, a loss of coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3c., above;~~
- 8) ~~Tube Inspection means an inspection of the steam generator tube from the point of entry tube end (hot leg side) completely around the U bend to the top support of the cold leg; and~~
- 9) ~~Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent Inservice inspections.~~

05-03A

- b. ~~The steam generator tube integrity shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through wall cracks) required by Table 4.4 2.~~

05-01A

05-02A

SURVEILLANCE REQUIREMENTS (continued)

4.4.5.5 Reports

- a. ~~Within 15 days following the completion of each Inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.~~
- b. ~~The complete results of the steam generator tube Inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:~~
 - 1) ~~Number and extent of tubes inspected,~~
 - 2) ~~Location and percent of wall thickness penetration for each indication of an imperfection, and~~
 - 3) ~~Identification of tubes plugged.~~
- c. ~~Results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

TABLE 4.4-1

05-017A

MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
	Two	Three	Four	Two	Three	Four
No. of Steam Generators per Unit						
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One ¹			One ¹	One ²	One ³

TABLE NOTATIONS

- ~~1. The Inservice inspection may be limited to one steam generator on a rotating schedule encompassing $3/N$ % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.~~
- ~~2. The other steam generator not inspected during the first Inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.~~
- ~~3. Each of the other two steam generators not inspected during the first Inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.~~

TABLE 4.4-2

05:01A

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	None	N.A.	N.A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
			C-3	Perform action for C-3 result of first sample	C-2	Plug defective tubes
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50	C-3	Perform action for C-3 result of first sample	C-3	Perform action for C-3 result of first sample
			All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50	N.A.	N.A.	N.A.	N.A.	

~~S = 3 N % Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection~~

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

- a. The Containment Atmosphere Particulate Radioactivity Monitoring System.
- b. The Containment Structure Sumps and the Reactor Cavity Sump Level and Flow Monitoring System, and
- c. Either the Containment Fan Cooler Collection Monitoring System or the Containment Atmosphere Gaseous Radioactivity Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

NOTE -
LCO 3.0.4 is not applicable

06-23-LS25

With only two of the above required Leakage Detection Systems OPERABLE, operation may continue for up to 30 days provided a RCS water inventory balance is performed* at least every 24 hours when the containment sump level and flow monitoring system is inoperable and either grab samples of the containment atmosphere are obtained and analyzed OR a RCS water inventory balance is performed* at least once per 24 hours when the required Gaseous and/or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

06-01-M

06-02-LS8

SURVEILLANCE REQUIREMENTS

4.4.6.1 The required Leakage Detection Systems shall be demonstrated OPERABLE by:

06-03-A

- a. Containment Atmosphere Particulate and Gaseous (if being used) Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.
- b. Containment Structure Sumps and the Reactor Cavity Sump Level and Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months, and
- c. Containment Fan Cooler Collection Monitoring System (if being used) -- performance of CHANNEL FUNCTIONAL TEST at least once per 18 months.

* Not required until 12 hours after establishment of steady state operation

06-26-LS30

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

- a. No PRESSURE BOUNDARY LEAKAGE,
- b. 1 gpm UNIDENTIFIED LEAKAGE,
- c. 1 gpm total reactor-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,
- d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,
- e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and,
- f. ~~1 gpm leakage at a Reactor Coolant System pressure of 2235 ± 20 psig for Reactor Coolant System Pressure Isolation Valves as specified in Table 3.4-1. Leakage from each Reactor Coolant System Pressure Isolation Valve shall be < 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure > 2215 psig and < 2255 psig.~~

06-06-A

06-07-LG

06-25-LS26

APPLICABILITY: MODES 1, 2, 3, and 4#.

06-08-LS9

ACTION:

- a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reactor coolant pump (RCP) seal injection flow, and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

06-09-LS10

~~(new) With RCP seal injection flow greater than the above limit, verify >100% flow equivalent to a single OPERABLE ECCS charging train is available within 4 hours and reduce the flow rate to within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.~~

06-21-LS35

06-09-LS10

- c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one two closed manual and/or deactivated automatic, or check valve##, and within 72 hours by the use of a second series closed manual, deactivated automatic, or check valve ##, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

06-11-LS11

06-12-M

REACTOR COOLANT SYSTEM

ACTION: (continued)

1) Pressure isolation valves in the residual heat removal (RHR) flow path when in, or during transition to or from, the RHR mode of operation are excluded in MODE 4.

06-08:LS9

Each valve used to satisfy this action must have been verified to meet surveillance requirement 4.4.6.2.2.

06-09:LS10

06-12:M

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. ~~Monitoring the containment atmosphere particulate or gaseous radioactivity monitor at least once per 12 hours;~~

06-13:LS12

b. ~~Monitoring the containment structure sump inventory and discharge at least once per 12 hours;~~

06-13:LS12

c. ~~a~~ Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals at least once per 31 days when the Reactor Coolant System pressure is 2235 + 20 psig with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

06-14:A

d. ~~b~~ Performance of a Reactor Coolant System water inventory balance at least once per 72 hours, ~~except when T_{avg} is being changed by greater than 5°F/hour or when diverting reactor coolant to the liquid holdup tank, in which cases the required inventory balance shall be performed within 12 hours after completion of the excepted operation; and not required to be performed until 12 hours after establishment of steady state operation~~

06-17:LG

e. ~~Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.~~

06-26:LS30

06-16:LS14

4.4.6.2.2 As specified in Table 3.4-1, Reactor Coolant System pressure isolation valves shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

06-07:LG

06-20:A

a. Every refueling outage during startup,

b. ~~Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and~~

06-19:TR3

c. ~~b~~ Within 24 hours following valve actuation due to automatic or manual action or flow through the valve. After each disturbance of the valve, in lieu of measuring leak rate, leak-tight integrity may be verified by absence of pressure buildup in the test line downstream of the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

TABLE 3.4.1

06-07.LG

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

<u>VALVE NUMBER</u>	<u>FUNCTION</u>
1. 8948 A, B, C, and D	Accumulator, RHR and SIS first off check valves from RCS cold legs
2. 8819 A, B, C, and D	SIS second off check valves from RCS cold legs
3. 8818 A, B, C, and D	RHR second off check valves from RCS cold legs
4. 8956 A, B, C, and D	Accumulator second off check valves from RCS cold legs
5. 8701* and 8702*	RHR suction isolation valves
6. 8949# A, B, C, and D.	RHR and SIS first off check valves from RCS hot legs
7. 8905# A, B, C, and D	SIS second off check valves from RCS hot legs
8. 8740# A and B	RHR second off check valves from RCS hot legs
9. 8802*# A and B	SIS to RCS hot legs isolation valves
10. 8703*#	RHR to RCS hot legs isolation valve

~~* Testing per Specification 4.4.6.2.2c. not required.~~

~~# For flow paths with 3 pressure isolation valves in series, at least 2 of the 3 valves shall meet the requirements of Specification 3.4.6.2f.~~

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REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

- a. Less than or equal to 1 microcurie/gram DOSE EQUIVALENT I-131, and
- b. Less than or equal to $100/\bar{E}$ microcuries/gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3*, ~~4, and 5.~~

08-01-LS16

ACTION:

~~MODES 1, 2, and 3*:~~

08-01-LS16

~~ECO 3.0.4 is not applicable for operation in ACTION a.~~

08-02-LS17

- a. With the specific activity of the reactor coolant greater than 1 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and
- b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microcuries/gram of gross radioactivity, be in at least HOT STANDBY ~~within 6 hours and reduce with T_{avg} to less than 500°F within the following 6 hours.~~

04-05-LS31

~~MODES 1, 2, 3, 4, and 5:~~

08-01-LS16

With the specific activity of the reactor coolant greater than 1 microcurie/gram DOSE EQUIVALENT I-131 ~~or greater than $100/\bar{E}$ microcuries/gram of gross radioactivity,~~ perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

08-01-LS16

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

* With T_{avg} greater than or equal to 500°F.

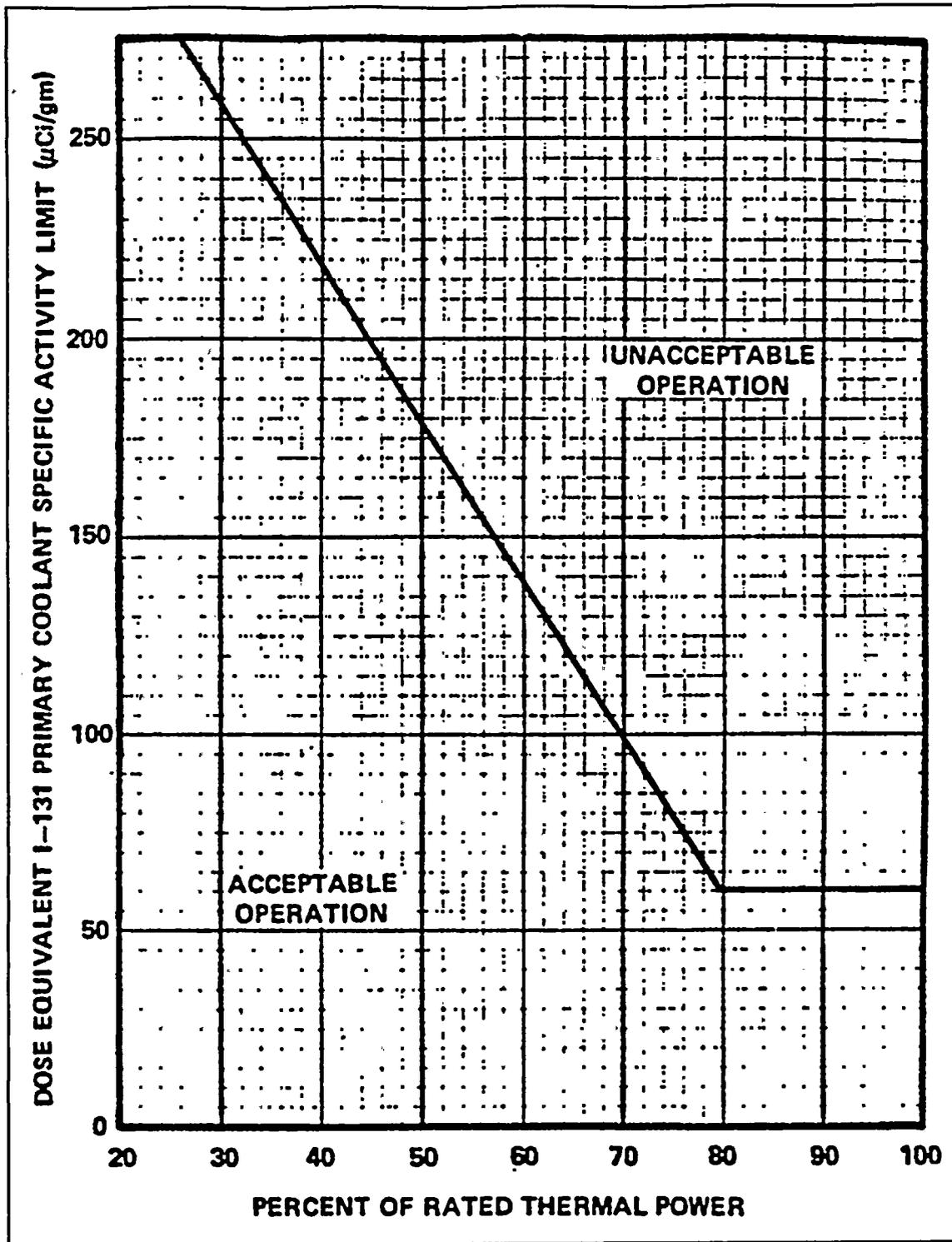


FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT
 VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT
 SPECIFIC ACTIVITY > 1 $\mu\text{CI/GRAM}$ DOSE EQUIVALENT I-131

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE
AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED	
1. Gross Radioactivity Determination**	At least once per 7 days 72 hours	1, 2, 3, with T_{avg} greater than or equal to 500° F. 4	<u>08-03-LS18</u> <u>08-01-LS16</u>
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1	
3. Radiochemical for E Determination***	1 per 6 months*	1	
4. Isotopic Analysis for DOSE EQUIVALENT I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 μ Ci/gram DOSE EQUIVALENT I-131 or 100/E μ Ci/gram of gross radioactivity, and	1#, 2#, 3# with T_{avg} greater than or equal to 500° F. 4#, 5#	<u>08-04-M</u> <u>08-01-LS16</u>
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3	<u>08-05-LS19</u>

TABLE 4.4-4 (Continued)

TABLE NOTATIONS

Until the specific activity of the Reactor Coolant System is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer. ~~Not required to be performed until 31 days after a minimum of 2 EFPDs and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for greater than 48 hours.~~

08-07-A

~~** A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides.~~

08-08-EG

*** A radiochemical analysis for E shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of E for the reactor coolant sample. Determination of the contributors to E shall be based upon those energy peaks identifiable with a 95% confidence level.

08-06-A

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit specified in the PRESSURE AND TEMPERATURE REPORT (PTER) lines shown on Figures 3.4.2 and 3.4.3 during heatup, cooldown, criticality, and Inservice leak and hydrostatic testing with:

09:01:LG

- a. ~~A maximum heatup of 100°F in any 1 hour period.~~
- b. ~~A maximum cooldown of 100°F in any 1 hour period.~~
- c. ~~A maximum temperature change of less than or equal to 10°F in any 1 hour period during Inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.~~

09:01:LG

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, in MODES 1, 2, 3, or 4, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation* within 72 hours, or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

09:02:M

09:03:M

~~With any of the above limits exceeded at any time in other than MODES 1, 2, 3, or 4, immediately initiate action to restore parameter(s) within limits and determine the RCS is acceptable for continued operation* prior to entering MODE 4.~~

09:02:M

SURVEILLANCE REQUIREMENTS

4.4.9.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per hour 30 minutes during system heatup, cooldown, and Inservice leak and hydrostatic testing operations.

09:16:M

~~*Acceptability for continued operation must be determined whenever the ACTION is entered.~~

09:02:M

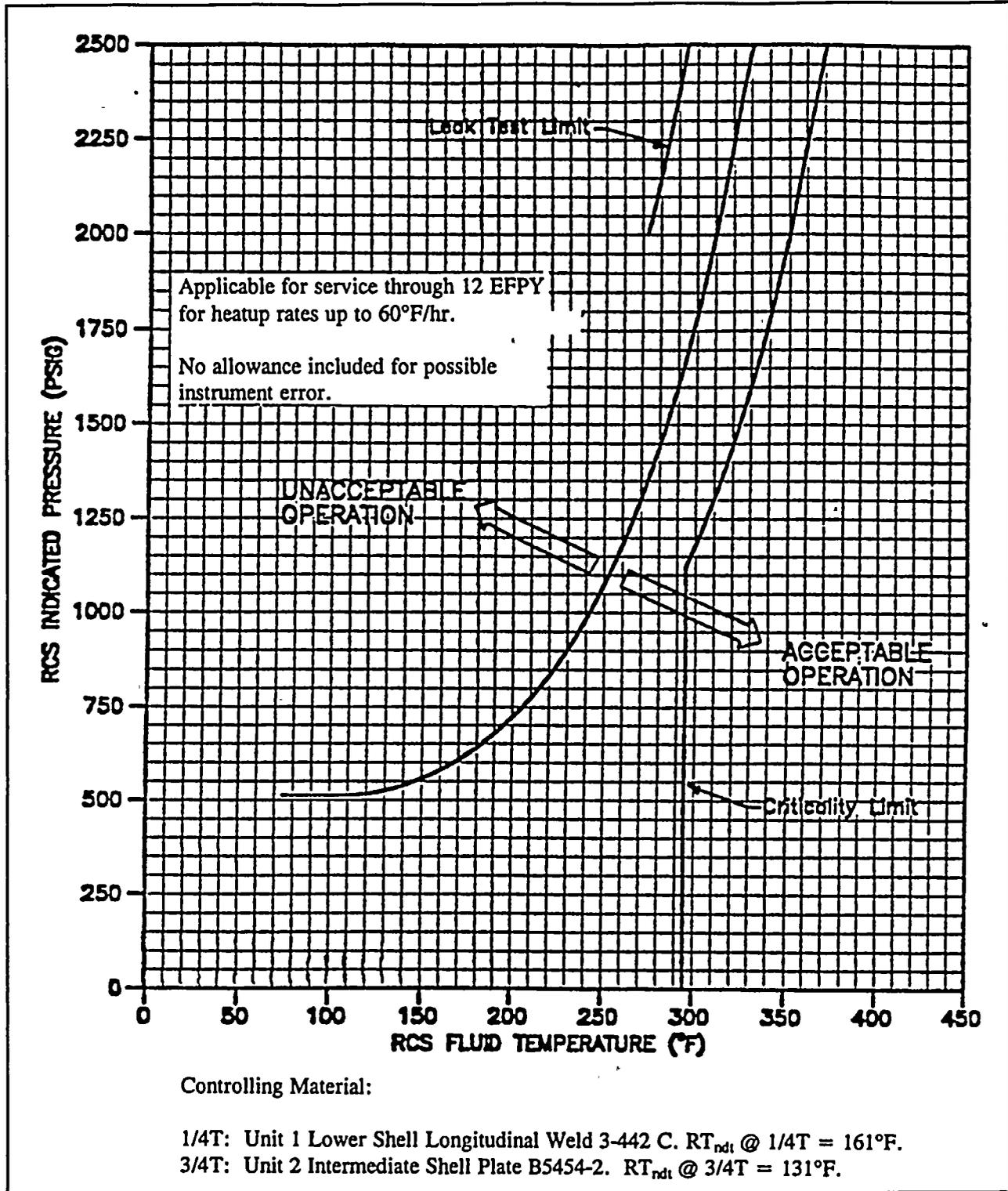


FIGURE 3-4-2
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 12 EFPY

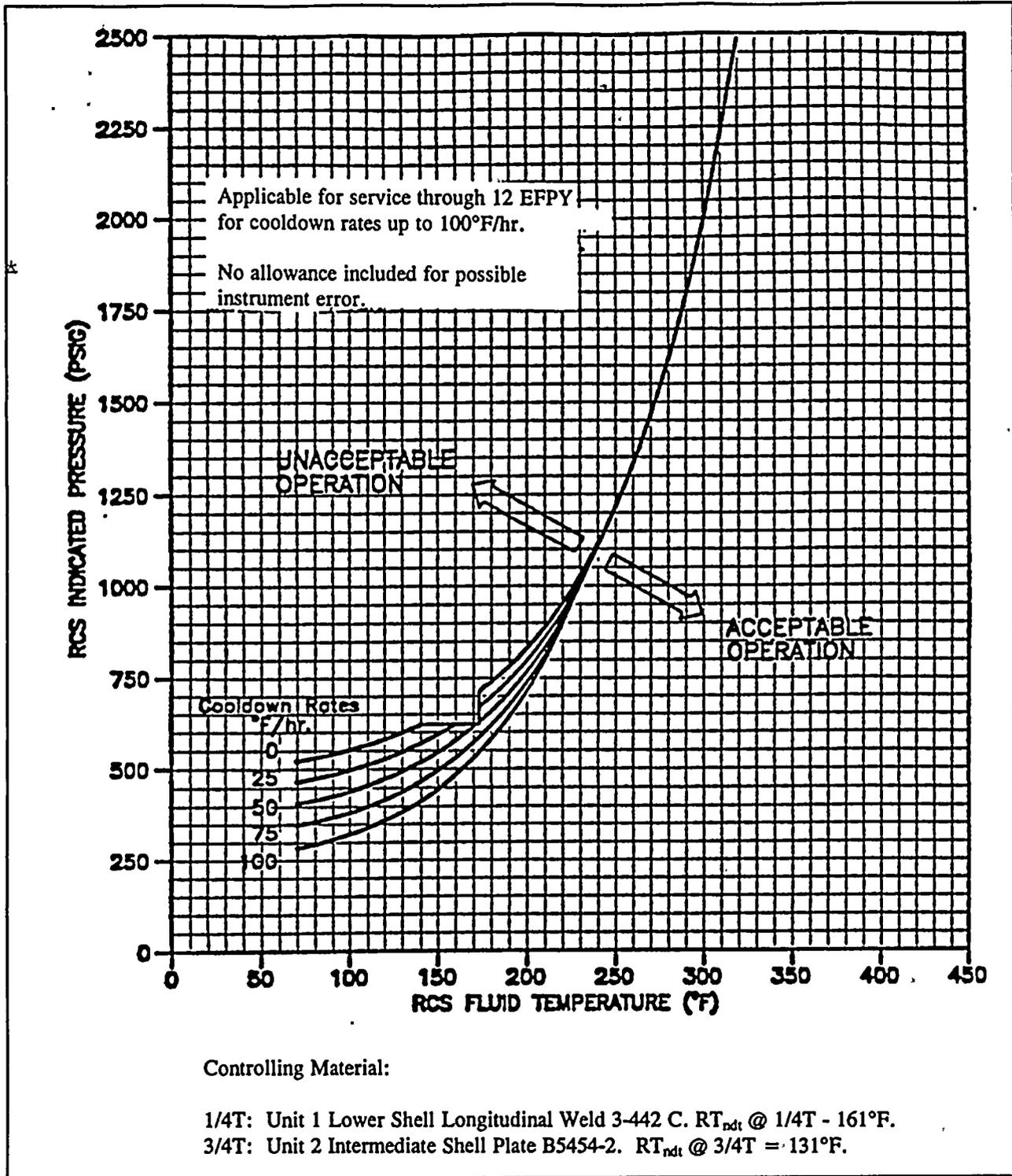


FIGURE 3.4.3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS — APPLICABLE UP TO 12 EFPY

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REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE with the accumulators isolated or depressurized below allowed RCS pressure per PTLR

09-06:M

- a. Two Class 1 power-operated relief valves (PORVs) with a lift setting of less than or equal to 435 psig, within the limits specified in the PTLR, or
- b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.07 square inches.

09-01:LG

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 270°F, the temperature below which LTOP is required as specified in the PTLR; MODE 5 and MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully de-tensioned.

01-17:LG

(new) The limitation for a maximum of one charging pump capable of injecting into the RCS is not required for pump swap operation until 1 hour after completion of pump swap operation.

09-08:LS7

ACTION:

- a. With one Class 1 PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches vent within the next 8 hours.
- b. With one Class 1 PORV inoperable in MODES 5 or 6 with the reactor vessel head on and the vessel head closure bolts not fully de-tensioned, restore the inoperable PORV to operable status within 24 hours or depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches within the next 8 hours.
- c. With both PORVs inoperable, depressurize and vent the RCS through an RCS vent of greater than or equal to 2.07 square inches vent within 8 hours.
- d. ~~In the event either the PORVs or the RCS vent are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent on the transient, and any corrective action necessary to prevent recurrence.~~

09-07:TR2

(new) With two charging pumps capable of injecting into the RCS, immediately initiate action to verify a maximum of one charging pump is capable of injecting into the RCS or within the next 8 hours, depressurize the RCS and establish RCS vent of greater than or equal to 2.07 square inches.

09-09:M

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION (continued)

ACTIONS (continued)

(new) The limitation for a maximum of one charging pump is capable of injecting into the RCS is not required for pump swap operation until 1 hour after completion of pump swap operation.	09:08:LS7
(new) With one or more safety injection pumps capable of injecting into the RCS, immediately initiate action to verify that a maximum of zero pumps are capable of injecting into the RCS.	09:15:M
(new) With an accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed by the P/T curves in the PTLR, within 1 hour isolate the affected accumulator or within the next 12 hours either increase all RCS cold leg temperatures to greater than the temperature below which LTOP is required as specified in the PTLR, or depressurize the affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed by the P/T curves in the PTLR.	09:10:M 01:17:LG
(new) With the LTOP system inoperable for any other reason, within 8 hours, depressurize the RCS and establish RCS vent of greater than or equal to 2.07 square inches.	09:11:M

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each Class 1 PORV shall be demonstrated OPERABLE by:

- a. Performance of a CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days; within 12 hours after entering MODE 4 when the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE. 09:12:LS20
- b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and
- c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent shall be verified to be open when the vent is being used for overpressure protection at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise verify the vent pathway every 12 hours.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION (continued)

(new) Verify that each accumulator is isolated at least once per 12 hours when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

09:14M

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Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. Deletions - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. Additions - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. Modifications - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. Administrative - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

**Methodology For Mark-Up of Current TS
(Continued)**

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers

(1 Page)

Description of Changes

(14 Pages)

TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS

SECTION 3/4.4

Technical Specification Title	CHG NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Reactor Coolant Loops Startup and Power Operations	1	3.4.1.1	3.4.1.1	3.4.1.1	3.4.1.1
Reactor Coolant Loops Hot Standby	1	3.4.1.2	3.4.1.2	3.4.1.2	3.4.1.2
Reactor Coolant Loops Hot Shutdown	1	3.4.1.3	3.4.1.3	3.4.1.3	3.4.1.3
Reactor Coolant Loops Cold Shutdown - Loops Filled	1	3.4.1.4.1	3.4.1.4.1	3.4.1.4.1	3.4.1.4.1
Reactor Coolant Loops Cold Shutdown - Loops Not Filled	1	3.4.1.4.2	3.4.1.4.2	3.4.1.4.2	3.4.1.4.2
Safety Valves Shutdown	2	N/A	N/A	3.4.2.1	N/A
Safety Valves Operating	2	3.4.2.2	3.4.2.2	3.4.2.2	3.4.2.2
Pressurizer	3	3.4.3	3.4.3	3.4.3	3.4.3
Relief Valves	4	3.4.4	3.4.4	3.4.4	3.4.4
Steam Generators	5	3.4.5	3.4.5	N/A	3.4.5
Reactor Coolant System Leakage - Leakage Detection Systems.	6	3.4.6.1	3.4.6.1	3.4.5.1	3.4.6.1
Reactor Coolant System Leakage - Operational Leakage	6	3.4.6.2	3.4.6.2	3.4.5.2	3.4.6.2
Chemistry	7	N/A	N/A	3.4.6	N/A
Specific Activity	8	3.4.8	3.4.8	3.4.7	3.4.8
Pressure/Temperature Limits - Reactor Coolant System	9	3.4.9.1	3.4.9.1	3.4.8.1	3.4.9.1
Pressure/Temperature Limits - Pressurizer	9	N/A	N/A	3.4.8.2	N/A
Pressure/Temperature Limits - Overpressure Protection Systems	9	3.4.9.3	3.4.9.3	3.4.8.3	3.4.9.3
Structural Integrity	10	N/A	N/A	3.4.9	N/A
Reactor Coolant System Vents	11	N/A	N/A	3.4.10	N/A

DESCRIPTION OF CHANGES TO TS SECTION 3/4.4

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant - specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant - specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	LG	The definition of OPERABLE and operating for the Reactor Coolant System (RCS) loops is moved to the Bases. Moving this information to the Bases is consistent with the NUREG-1431 philosophy of moving clarifying information and descriptive details out of the TS to the Bases.
01-02	A	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-03	LS1	This change adds the additional allowance that the Rod Control System is capable of rod withdrawal before the ACTION is applied. This is a relaxation because methods, other than opening the Reactor Trip Breakers (RTBs), are allowed to preclude rod withdrawal. However, as prevention of rod withdrawal is the requirement of concern, the relaxation is justified. This change is in conformance with NUREG-1431. The Limiting Condition for Operation (LCO) is also revised to be based on capability to withdraw rods rather than RTB position.
01-04	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B). This test exception is not in the current TS.
01-05	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-06	M	The Note to allow short term de-energization of Reactor Coolant Pumps (RCPs) and Residual Heat Removal (RHR) pumps in MODES 3, 4, and 5, for 1 hour is clarified as 1 hour per 8 hour period in CTS LCOs 3.4.1.2, 3.4.1.3, and 3.4.1.4.1. This change is in accordance with NUREG-1431 as modified by Industry Traveler TSTF-153 and is more restrictive as no restrictions on use of the Note were required previously.
01-07	M	This change separates the ACTION Statement for one loop OPERABLE from that of no loops OPERABLE. The condition of no loops OPERABLE is now included with the condition of no loops in operation. The additional ACTION of returning a loop to OPERABLE status is added for this change. The effect of this change is more conservative operation as a result of no loops being OPERABLE. This change is consistent with present plant operations. This change is in conformance with NUREG-1431.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.4

CHANGE NUMBER

NSHC

DESCRIPTION

01-08	LS2	This change adds a specific relaxation to allow the use of an operating RCS loop in lieu of an operating RHR loop in MODE 5 during planned heatup in preparation to proceed into MODE 4. The primary functions of the operating RHR loop in MODE 5 are to remove decay heat and to prevent boron stratification. These functions can also be performed by an operating RCS loop which is a normal method of accomplishing the same functions when in MODE 4. In addition, at least one RHR loop must remain OPERABLE during the transition to MODE 4. The change is acceptable because there is no reduction in the heat removal/boron stratification capability or system reliability when the RCS loop is performing these functions. This change is in conformance with NUREG-1431.
01-09	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-10	M	The ACTION is changed to include the Required Actions for no required RHR loops OPERABLE. Previously, the specification did not provide specific ACTION for no RHR loops OPERABLE. This change also clarifies that the restored loop must be OPERABLE as well as in operation. This ACTION is consistent with the ACTIONS which are required under this Condition and are in conformance with NUREG-1431.
01-11	M	This change adds a new surveillance for verification of breaker alignment and power availability to the required pump that is not in operation. This change is in conformance with NUREG-1431.
01-12	M	The ACTIONS are changed to delineate the Required Actions for only one required RHR loop OPERABLE. These revised ACTIONS are consistent with the ACTIONS which are required under this LCO in conformance with NUREG-1431 and are more conservative than current Required Actions.
01-13	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-14	LS22	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-15	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-16	A	Consistent with the intent of Industry Traveler TSTF-153, this change revises the Note that permits up to 1 hour "deenergization" of RCP/RHR pumps. The revised wording clarifies the intent of the Note to allow the pumps to be "removed from operation" instead of "deenergized", thus permitting other means of removing the pumps from service. With this change the pumps are not <u>required</u> to be deenergized to use the Note (e.g., the pumps may be isolated, etc.). The change is considered to be administrative because from the standpoint of providing an exception to the LCO requirements (to maintain the operability and operation of the pumps), the revised wording is equivalent.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.4

CHANGE NUMBER

NSHC

DESCRIPTION

01-17	LG	The plant specific minimum temperature specified in degrees below which the RCS must not be subject to overpressure is replaced by the generic statement "the minimum low temperature value specified in the PRESSURE TEMPERATURE LIMITS REPORT (PTLR)". The minimum temperature is a plant specific requirement based on the reactor vessel material characteristics documented in the PTLR and is periodically reviewed and adjusted as required. Referring to the PTLR for the current value is consistent with the relocation of the pressure/temperature limits from improved TS Section 3.4.3 to the PTLR. This is an administrative change. In the CTS, this change involves the footnote for LCO 3.4.1.3, the footnote for Applicability 3.4.1.4.1, the Applicability and ACTION a of 3.4.2.2, the Applicability for 3.4.9.3, and ACTION (new) of 3.4.9.3.
01-18		Not used.
01-19	M	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-20	LS27	The restriction in the LCO footnote for reactor vessel water level above the reactor flange for the stopping of all RHR pumps is removed . Footnote condition 3) is replaced by the restriction to allow no draining operations that would reduce RCS water inventory. This revised footnote is less restrictive because the water inventory would be less with the level lower than the vessel flange. This change is in conformance with NUREG-1431.
02-01	LS3	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-02	M	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-03	M	This change in accordance with NUREG-1431 provides a new ACTION for two or more pressurizer safety valves inoperable. Prior to this change, LCO 3.0.3 would have been entered allowing for 1 hour prior to placing the unit in HOT STANDBY within the next 6 hours, etc. This change is more restrictive in that the 1 hour time frame is eliminated. This change is acceptable based on the analysis and commensurate actions required with two or more pressurizer safeties inoperable.
02-04	LG	The footnote about the lift setting being for ambient conditions is moved to the TS Bases. Moving this information to the Bases is consistent with the NUREG-1431 philosophy of moving clarifying information and descriptive details out of the TS to the Bases.
02-05	LS23	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

DESCRIPTION OF CHANGES TO TS SECTION 3/4.4

CHANGE NUMBER

NSHC

DESCRIPTION

03-01	LS4	<p>In conformance with NUREG-1431 the ACTION has been changed from the "pressurizer otherwise inoperable" to "water level not within limit." This change is slightly less restrictive because if the pressurizer was declared inoperable for reasons other than those listed in the actions, LCO 3.0.3 would be entered. LCO 3.0.3 entry would provide one additional hour beyond the completion times specified in the LCO. This change is acceptable because a high pressurizer water level is the primary condition in MODES 1, 2, and 3 which could lead to pressurizer inoperability and therefore requires the most immediate actions. High pressurizer level is potentially indicative of a loss of the pressurizer bubble which could adversely effect RCS pressure transients. The actions required for high pressurizer level are unaffected by this change. Any other cause of inoperability is not likely to require immediate action and is adequately protected by LCO 3.0.3.</p>
03-02	LG	<p>The method of verifying heater capacity is moved to the Bases. Moving this information to the Bases is consistent with the NUREG-1431 philosophy of moving clarifying information and descriptive details out of the TS to the Bases.</p>
03-03	LS28	<p>This change relaxes the surveillance requirement for verifying the capacity of the pressurizer backup heaters from 92 days to 18 months. The OPERABILITY of the pressurizer heaters enhances the capability of the plant to control RCS pressure and establish natural circulation. The purpose of the surveillance requirement is to detect potential pressurizer heater degradation. This is done by periodically demonstrating that pressurizer heaters are capable of producing power at their design rating, by testing the power supply output and by performing electrical checks on heater elements. Heating elements are simple resistive devices which are not prone to complex failure modes. Moreover, heater banks are made up of a number of individually powered heater elements such that a common mode failure of the elements is unlikely. The normal power supply to the heaters is in use during normal power operations and a failure of the power supply would be immediately detectable independent of the heater surveillance. The low failure rate experience with pressurizer heater elements is indicative that the surveillance interval may be extended without loss in heater reliability. This change is consistent with Generic Letter 93-05, "Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation" and Industry Traveler TSTF-93, Rev. 1.</p>
03-04	LS29	<p>This ACTION is revised to reflect its LCO. The position of the reactor trip breakers is not an LCO requirement, therefore the ACTION is revised. As currently worded, the ACTION requires the reactor trip breakers to be open. This position could preclude certain testing in MODE 3. A more generic ACTION statement which assures the rods are fully inserted and can not be withdrawn, replaces the specific method of precluding rod withdrawal. The revised ACTION still assures rod withdrawal is precluded, and details of method are not required to be in the TS to provide adequate protection of the public safety. No technical changes result from this revision. This change is based on Industry Traveler TSTF-87, Rev. 1.</p>

DESCRIPTION OF CHANGES TO TS SECTION 3/4.4

CHANGE NUMBER

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DESCRIPTION

03-05	M	This change adds a requirement to CTS 3.4.3. The change requires the pressurizer heaters to be capable of being powered from an emergency power source. This change, in accordance with NUREG-1431, is not in the CTS, and is more restrictive.
04-01	LS5	In conformance with NUREG-1431, an allowance has been added that separate condition entry is allowed for each valve. This allowance is acceptable because the functionality of each pressurizer power operated relief valve (PORV) is addressed separately within the specification and the allowed Completion Times are based on the discovery of inoperability for each PORV/block valve combination.
04-02	LS6	In conformance with NUREG-1431, the allowance for PORV inoperability has been expanded from previously only allowing block valve isolation when the PORV was inoperable for seat leakage to allowing block valve isolation when the PORV is still capable of manually being cycled. With the PORV still capable of being manually cycled it is available for mitigation of accidents such as the steam generator tube rupture (SGTR) but has lost the functionality of automatic pressure reduction. As loss of the automatic function does not result in a degradation of equipment which is required to operate during an accident, this change is acceptable.
04-03	M	The times to reduce MODE for inoperable PORVs are decreased by one hour. This change is acceptable because it is a more restrictive change.
04-04	LG	The automatic actuation of the PORVs is not credited in MODES 1, 2, or 3. Therefore, performance of the channel calibration is unnecessary as a TS requirement. Consequently, the channel calibration requirement is moved to a licensee controlled document. The portions of the PORV actuation circuitry if required for [LTOP] are calibrated in accordance with ITS SR 3.4.12.9. Moving this information is consistent with the NUREG-1431 philosophy of moving requirements, which are not required to satisfy accident analyses, out of the TS.
04-05	LS31	The shutdown requirement of CTS 3.4.4 would require the plant to reduce T_{avg} to <500°F within 12 hours, rather than go to MODE 4, to address the concern of entering [LTOP] LCO Applicability with inoperable PORVs. For consistency, the shutdown requirements of CTS LCO [3.4.8] would be similarly revised. The proposed change adds a relaxation to the ACTIONS associated with CTS LCO 3.4.4 by keeping the end point of the shutdown action above the LCO [3.4.9.3] Applicability. No credit is taken for the automatic actuation of the PORVs in MODES 1, 2, or 3. Credit is taken for the manual operation of the PORVs during a SGTR. The capability to manually cycle the PORVs will be unaffected by this change. The release of radioactivity in the event of an SGTR with RCS T_{avg} <500°F is unlikely since the saturation pressure of the reactor coolant would be less than the lift pressure of the main steam safety and [SG atmospheric relief] valves. This change is in accordance with Industry Traveler TSTF-113.
04-06	LS32	This change provides a 72 hour completion time to restore an inoperable block valve, with the PORV placed in manual control mode. The CTS requires the block valve to be restored within one hour or remove power from the PORV solenoid. This change is in accordance with NUREG-1431 and is less restrictive.

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04-07	LS33	This change provides a two hour completion time for restoring an inoperable block valve when more than one block valve is inoperable, and 72 hours to restore the remaining valves. The CTS requires the inoperable block valves to be restored within one hour for one or more valves inoperable. This change is in accordance with NUREG-1431 and is less restrictive.
04-08	LS34	Consistent with Industry Traveler WOG-60, the requirement to perform the 92 day surveillance of the pressurizer PORV block valves and the 18 month surveillance of the pressurizer PORVs (i.e., perform one complete cycle of each valve is revised to indicate that the surveillance is only required to be performed in MODES 1 and 2. This footnote allows entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. This is consistent with the recommendations of Generic Letter 90-06, 'Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10CFR50.54(f), June 25, 1990., which indicates that administrative controls require this test be performed in MODES 3 or 4 to adequately simulate operating temperature and pressure effects on valvePORV operation.
4-09	LS36	The requirement to perform the 92 day surveillance of pressurizer PORV block valves (i.e., perform one complete cycle of each block valve) is revised such that it is not required if the block valve is closed to meet ACTION a of CTS LCO 3.4.4. This change is acceptable because no credit is taken for the automatic actuation of the PORV in MODES 1, 2, or 3. Credit is taken for manual operation of the PORVs during the SGTR accident. However, the capability to manually cycle the PORVs will be unaffected by this change. This change will not affect the ability of the block valve to open; if closed to meet ACTION a, in the mitigation of an SGTR. Deferral of the block valve cycling surveillance will not diminish the design capability of the block valve to open against differential pressures that would be present after an SGTR since the block valves are capable of opening against 2485 psig, the safety valve lift pressure, whereas pressurizer pressure decreases after an SGTR. This change is consistent with traveler WOG-87.
05-01	A	The SG tube surveillance requirements are initiated by ITS SR 3.4.13.2 and the specifics are moved to ITS Administrative Controls Sections 5.5.9, "Steam Generator (SG) Tube Surveillance Program" and 5.6.10, "Steam Generator Tube Inspection Report."
05-02	A	CTS 3.4.5 is deleted for consistency with NUREG-1431. SG operability requirements in MODES 1 through 4 are specified in the RCS loop and operational LEAKAGE specifications.

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DESCRIPTION

05-03	A	<p>CTS SR 4.4.5.4.a.8) is clarified to remove potential interpretation problems related to probe orientation versus entry point. The TS specifies that a SG "tube inspection means an inspection for the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg." This statement is interpreted as defining the minimum portion of the tube requiring eddy current testing to meet surveillance requirements, not a particular eddy current probe orientation or entry point. This specification is based on Regulatory Guide 1.83, position c.2.f which requires that "The equipment should be capable of examining the entire length of the tubes." Footnote 3 of position c.2.f specifies for U-bends designed steam generators that an examination from the hot leg side completely around the U-bend to the top support of the cold leg is sufficient for this inspection but contains an inference to probe entry on the hot-leg side. Subsequent to Regulatory Guide 1.83 and development of the Standard Technical Specification provisions, Generic Letter 85-02 recommended that "tube inspections should include an inspection of the entire length of the tube (tube end to tube end) including the hot-leg side, U-bend, and cold-leg side." This recommendation is consistent with the phrasing of CTS SR 4.4.5.4.a.9) for Preservice Inspection and the Technical Specification for non-U-tube Steam Generators (B&W). These requirements do not infer any particular orientation during full length (tube end to tube end) inspections and, as stated in Generic Letter 85-02, should not preclude separate hot leg and cold leg sides of the same tube from being inspected to meet minimum sampling requirements.</p>
06-01	M	<p>Consistent with NUREG-1431, this change adds the requirement to perform an RCS water inventory balance every 24 hours when the [sump level detector] is inoperable. This is a new requirement and is more restrictive than the CTS.</p>
06-02	LS 8	<p>Consistent with NUREG-1431, this change allows the performance of an RCS water inventory balance every 24 hours as an alternative to the requirement to perform 24 hour samples of the containment atmosphere when a required radioactivity monitor is inoperable. The primary function of the leakage detection systems is to detect significant reactor coolant pressure boundary (RCPB) degradation as soon as practical to minimize the potential for propagation to a gross failure. The TS requires multiple diverse systems to ensure that leakage from a variety of locations and leakage rates can be detected in sufficient time to take measures to place the plant in a safe condition. When a required radioactivity monitor is inoperable, the CTS provides compensatory action by requiring containment atmosphere samples be taken and analyzed every 24 hours to supplement indications from detection systems which are still operable [(e.g., sump level and condensate flow rate)]. Another diverse means for determining RCS leakage is the RCS water inventory balance. When performed at the same frequency as the containment atmosphere grab sample, and in conjunction with other leak detection indications, this method provides an equal degree of confidence of the status of RCS leakage.</p>
06-03	A	<p>Consistent with NUREG-1431, this change adds the word "required" to clarify that only those detectors which are being used to satisfy the LCO must be demonstrated to be operable.</p>

DESCRIPTION OF CHANGES TO TS SECTION 3/4.4

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
06-04	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
06-05	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
06-06	A	Moves the LCO for CONTROLLED LEAKAGE (seal injection flow) from "Operational Leakage" to LCO 3.5.5 "ECCS" in conformance with NUREG-1431. The reference operating point for surveillance of 40 gpm controlled leakage is unchanged from nominal operating RCS pressure (+/- 20 psig) with the charging flow control valve full open which is moved to SR 3.5.5.1 in the improved TS.
06-07	LG	This change moves the listing of RCS pressure isolation valves (PIVs). This listing is being moved to a licensee controlled document as identified in the Conversion Comparison Table (Enclosure 3B). Moving this information is consistent with the NUREG-1431 philosophy of moving clarifying information and descriptive details out of the TS. The requirement to verify PIV leakage is moved to SR 3.4.14.1 in the ITS.
06-08	LS9	This change, in conformance with NUREG-1431, adds clarification that the valves in the RHR flow path are not required to meet the Pressure Isolation Valve specification "when in, or during the transition to or from the RHR mode of operation" in MODE 4. This change is acceptable because the same allowance was available in the current TS by an exception to Specification 4.0.4. In addition, using RHR through these specific flow paths is the required mode of operation during an accident.
06-09	LS10	The LCO applicability for CONTROLLED LEAKAGE (seal injection flow) is reduced to only MODES 1, 2, and 3 (with the associated change in ACTION b). This change is consistent with NUREG-1431. This change is acceptable because the limitation on this parameter is only required in MODES 1, 2, and 3 to support the LOCA analyses by ensuring sufficient ECCS flow. The limitation does not apply in MODE 4. Thus the shutdown requirement in ACTION b) is revised to require hot shutdown instead of cold shutdown.
06-10	LG	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
06-11	LS11	This change in conformance with NUREG-1431, allows for the flow path to be isolated by one valve within 4 hours and by a second series valve within 72 hours. This change is less restrictive and is acceptable based on the first valve having been surveilled as meeting the same leakage criteria as the inoperable PIV and the small probability of a failure during the 72 hour period. This action additionally allows a check valve to be used for isolation purposes when a PIV is inoperable. This change is less restrictive and is acceptable because the check valve in use has been surveilled in accordance with the same leakage criteria as the PIV.
06-12	M	Any valve(s) used to satisfy the action of isolating the low pressure system from the high pressure system when a RCS PIV is inoperable, must be tested and meet the leakage criteria of the PIV. This change is in conformance with NUREG-1431 and is more restrictive.

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06-13	LS12	This change in conformance with NUREG-1431, removes the specific requirement to monitor the RCS Leakage Detection System once per 12 hours. However, the change is acceptable because ITS SR 3.4.15.1 requires a CHANNEL CHECK be performed on the containment radioactivity monitor channels on the same frequency so there is no change in requirement for those detectors. The [containment sump level detection system] and the [containment fan cooling unit condensate collection monitor] can be monitored from the control room. [].
06-14	A	This change moves the CTS Surveillance for CONTROLLED LEAKAGE (seal injection flow) from "Operational Leakage" to ITS 3.5.5,"ECCS" in conformance with NUREG-1431.
06-15	LS13	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
06-16	LS14	This change in conformance with NUREG-1431, removes the requirement for monitoring the reactor head flange leakoff system. This change is acceptable because reactor head leakage, which is collected in the reactor coolant drain tank, is quantified as identified LEAKAGE. Identified LEAKAGE is determined by performance of a RCS water inventory balance. The initial inventory balance is required within 12 hours following RCS steady state operation and every 72 hours thereafter. Flange leakoff does not provide an indicator of pressure boundary integrity and consequently the removal of the monitoring does not involve the probability of loss of RCS water inventory.
06-17	LG	The definition of steady state is moved to the Bases. Moving this information is consistent with the NUREG-1431 philosophy of moving clarifying information and descriptive details out of the TS.
06-18	LS15	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
06-19	TR3	This change in conformance with NUREG-1431, removes the specific requirement for performing the PIV surveillance prior to returning a valve to service following maintenance, repair or replacement. This requirement is inherent to post maintenance operability requirements and removal from the specifications is not a reduction in testing scope or frequency.
06-20	A	Consistent with NUREG-1431, Inservice Testing (IST) requirements are moved to ITS 5.5.8.

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NSHC

DESCRIPTION

06-21	LS35	<p>This change increases the RCP seal injection flow Completion Time from 4 to 72 hours, with a new added verification that at least 100% of the assumed charging flow remains available. The Bases for the seal injection flow limit relates to ensuring adequate charging flow during post-LOCA injection. The revised ACTIONS continue to assure this basis is adequately addressed by providing an ECCS-like Required Action. ITS Specification 3.5.2 allows a 72 hour Completion Time for one or more ECCS subsystems inoperable if at least 100% of the assumed ECCS flow is available. The seal injection flow ACTIONS have been modified so that if the remaining charging flow (with some inoperability in the charging system) is greater than or equal to 100% of the assumed post-LOCA charging flow, 72 hours is allowed to restore OPERABILITY. This change is consistent with Industry Traveler WOG-84.</p>
06-22	M	<p>Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).</p>
06-23	LS25	<p>CTS 3.4.6.1, "Leakage Detection Systems", is revised such that the provisions of Specification 3.0.4 are not applicable. This will allow entry into the applicable MODES with only one of the Leakage Detection Systems OPERABLE, subject to the requirements of the ACTION statements. This change is consistent with NUREG-1431 and Industry Traveler TSTF-60 and is acceptable because of the diverse means available to detect RCS leakage.</p>
06-24	M	<p>Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).</p>
06-25	LS26	<p>The Operational Leakage LCO has been modified to change the allowed leakage limit for RCS PIVs for consistency with improved TS SR 3.4.14.1. The RCS PIV LCO permits system operation in the presence of leakage through valves in amounts that do not compromise safety.</p>
06-26	LS30	<p>The CTS surveillance requirement for performing a RCS water inventory balance is modified to allow deferral of the water inventory balance such that it would be performed within 12 hours after achieving steady state conditions. The RCS water inventory balance must be performed with the reactor at steady state conditions as discussed in the ITS Bases. This change is in conformance with Industry Traveler TSTF 116, Rev. 1.</p>
06-27	A	<p>Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).</p>
06-28	LG	<p>Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).</p>
07-01	R	<p>Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).</p>

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CHANGE NUMBER

NSHC

DESCRIPTION

08-01	LS16	<p>This change in conformance with NUREG-1431, revises the Applicability of the specification to MODES 1, 2, or 3 with $(T_{avg}) \geq 500^{\circ}\text{F}$. The change deletes the requirement to perform an isotopic analysis for Iodine every 4 hours in MODES 4 and 5 and in MODE 3 below 500°F, whenever the reactor coolant exceeds its DOSE EQUIVALENT IODINE limit. This change deletes the requirement to perform the once per 4 hour surveillance for DOSE EQUIVALENT I-131 in the event the gross specific activity limit is exceeded, in accordance with Industry Traveler TSTF-28. Further, this is an unnecessary requirement since the ACTION requires the plant to exit the LCO's revised Applicability. This change is acceptable as offsite release of radioactivity in the event of a SGTR is unlikely for operation below 500°F, as the saturation pressure of the reactor coolant is below the lift pressure of the settings of the main steam safety and [relief] valves.</p>
08-02	LS17	<p>This change in conformance with NUREG-1431, adds an exception to LCO 3.0.4 when operating in Condition A. This would allow MODE changes under conditions that the plant is anticipating a return to acceptable activity levels within the allowed Completion Time of Required Action A. 2. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.</p>
08-03	LS18	<p>This change in conformance with NUREG-1431, revises the sample frequency from 72 hours to 7 days for performance of a gamma isotopic analysis. The 7 day frequency is acceptable based on the low probability of a gross fuel failure occurring which would significantly alter the analysis results.</p>
08-04	M	<p>Consistent with NUREG-1431, the CTS requirement to measure Iodine including I-131, I-133 and I-135, is replaced with a requirement to measure a DOSE EQUIVALENT I-131. This is more restrictive.</p>
08-05	LS19	<p>This change in conformance with NUREG-1431, revises the performance of the surveillance of the specific activity of the RCS following a 15% power change to MODE 1 only. Previously the surveillance was also required if the plant was in MODES 2 or 3 which required the plant to perform this surveillance following a Reactor Trip when normal spiking would be anticipated and the results would not be indicative of the operational status of the RCS.</p>
08-06	A	<p>The definition of E-bar is located in the Definition Section of NUREG-1431 and is not duplicated in the corresponding ITS specification (ITS 3.4.16.)</p>
08-07	A	<p>Consistent with NUREG-1431, a Note was added to clarify the time in which the sample is required to be taken upon meeting the minimum conditions to obtain the sample. This is administrative in nature as it is added for clarification.</p>
08-08	LG	<p>The description of gross radioactivity analysis is moved to the ITS Bases in conformance with NUREG-1431.</p>

DESCRIPTION OF CHANGES TO TS SECTION 3/4.4

CHANGE NUMBER

NSHC

DESCRIPTION

09-01	LG	In conformance with NUREG-1431, the CTS figures which contain operational parameters related to the RCS Pressure/Temperature (P/T) Limits are being moved to the RCS PTLR. The PTLR will be in accordance with ITS 5.6.6.
09-02	M	In conformance with NUREG-1431, the ACTIONS are broken into MODES 1, 2, 3, or 4 and a separate ACTION for "at all other times." The MODES 1, 2, 3, or 4 ACTION is essentially the same as the previous ACTIONS. The ACTION which covers other times requires immediate action to restore the parameters and a determination of RCS acceptability prior to entering MODE 4. Both ACTIONS are modified to require a determination of acceptability for continued operation whenever the ACTIONS are entered. These ACTIONS are more restrictive but are acceptable.
09-03	M	In conformance with NUREG-1431, the time allowed for performance of the engineering evaluation following exceeding the P/T limits has been defined as 72 hours. No specific time limit was provided in the current specifications except for the 30 minutes allotted to restore the parameters to within limits. 72 hours is reasonable based on the necessary time for performing the evaluation and the low probability of the RCS structural integrity being unacceptable.
09-04	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
09-05	R	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
09-06	M	In conformance with NUREG-1431, the requirement is included to either isolate the accumulators or depressurize them to a pressure corresponding with the RCS temperature to preclude an RCS overpressure event. This requirement is commensurate with current practice and although addition to the improved TS is more restrictive, the change is necessary to help ensure [LTOP] analysis assumptions are maintained.
09-07	TR 2	This change in accordance with NUREG-1431 removes the requirement for a special report to be generated and submitted to the NRC following the mitigation of an RCS pressure transient. Reporting to the NRC will be done commensurate with the reporting requirements of 10 CFR 50.72 and 50.73.
09-08	LS7	A Note is added to CTS 3.4.9.3 ACTION to allow two centrifugal charging pumps (CCP)s capable of injecting into the RCS during pump swap operation for \leq 1 hour. The [LTOP] system design requires no more than one CCP capable of injection. This change provides flexibility to switch operating pumps without interrupting injection flow. This is in accordance with NUREG-1431 and Industry Traveler WOG-51 and is a less restrictive change.
09-09	M	In conformance with NUREG-1431 an additional ACTION is added for [two] charging pumps capable of injecting into the RCS and subsequent actions for RCS depressurization and venting if necessary.

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CHANGE NUMBER

NSHC

DESCRIPTION

09-10	M	In conformance with NUREG-1431 an additional ACTION is added for an accumulator which has not been isolated or depressurized and subsequent actions for RCS depressurization and venting if necessary.
09-11	M	In conformance with NUREG-1431 an additional ACTION is added for the system being inoperable for any other reason to include ACTION for RCS depressurization and venting. The addition of this ACTION is a more restrictive change which is necessary to help ensure [LTOP] analysis assumptions are maintained.
09-12	LS20	In conformance with NUREG-1431 an allowance has been included which provides for performance of the CHANNEL OPERATIONAL TEST (COT) on the PORV actuation channels within 12 hours after entering the applicability. Previously the COT was required to be performed prior to entry and was an unnecessary burden on plant operation. As stated in the improved TS BASES, the 12 hour time frame is an acceptable period for performance of the COT. This change is less restrictive and is acceptable as stated above.
09-13	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
09-14	M	In conformance with NUREG-1431 an additional surveillance is added to verify accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. This change is consistent with adding the requirement for accumulator isolation to the LCO (see CN 09-06-M). The addition of this surveillance requirement is a more restrictive change which is necessary to help ensure [LTOP] assumptions are maintained.
09-15	M	In conformance with NUREG-1431, an additional ACTION and corresponding SR are added for one or more Safety Injection (SI) pumps or more than one CCP capable of injection into the RCS and subsequent actions for RCS depressurization and venting if necessary. [These items are in the current ECCS TS.]
09-16	M	CTS surveillance 4.4.9.1 determines RCS temperature and pressure are within limits once per hour during RCS heatup, cooldown, or testing. ITS SR 3.4.3.1 requires verification of RCS conditions within limits once per 30 minutes. This is a more restrictive change.
09-17	LS24	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
10-01	LS21	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

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10-02

A

Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

10-03

LS37

Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

11-01

R

Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(15 pages)

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01-LG	The definition of OPERABLE and operating for the RCS loops is moved to the Bases.	Yes	Yes	Yes	Yes
01-02-A	Surveillances to verify steam generator OPERABILITY are moved to SR 3.4.13.2 and Section 5.5.9 in the improved TS.	No, DCPD does not have this SR in 3/4.4.1.	Yes	No, WCGS does not have this SR in 3/4.4.1.	No, Callaway does not have this SR in 3/4.4.1.
01-03-LS1	This change replaces the specific requirement that RTBs be open with the more general requirement that the rod control system be not capable of rod withdrawal.	Yes	Yes	No, WCGS did not have the specific requirement that the RTBs be open. See change 1-14-LS.	No, Callaway did not have the specific requirement that the RTBs be open. See change 1-14-LS.
01-04-M	This change removes the special test exception (3.10.4) which allowed for suspending this LCO during the performance of hot rod drop time measurements.	No, not in the DCPD CTS.	Yes	Yes	Yes
01-05-M	Notes are added or revised in current TS LCOs 3.4.1.2, 3.4.1.3, 3.4.1.4.1 to reflect [COMS] analysis requirements restricting conditions for starting an idle RCP.	No, not in CTS	No, not in CTS	Yes	Yes
01-06-M	The allowed short term de-energization of RCPs for 1 hour (with no restrictions) is changed to 1 hour per 8 hour period.	Yes	Yes	Yes	Yes
01-07-M	This change separates the ACTION Statement for one loop OPERABLE from that of no loops OPERABLE. The condition of no loops OPERABLE is now included with the condition of no loops in operation. The additional ACTION of returning a loop to OPERABLE status is added.	Yes	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-08-LS2	This change adds an allowance to suspend operation of all RHR pumps when proceeding from MODE 5 to MODE 4, if decay heat removal is being provided by an RCS loop. This change is to will provide a smoother transition to MODE 4 operations.	Yes	Yes	Yes	Yes
01-09A	This change replaces the word "or" with "and". The ACTION could have been interpreted to require that the SG level be restored even when the SGs were not being credited for heat removal. The revised wording clarifies that the second part of the LCO (i.e., 'a' or 'b') is met with either one RHR loop OPERABLE or two SGs with required level.	No, DCPD CTS already has an "and."	Yes	No, WCGS CTS already has an "and".	No, Callaway CTS already has an "and".
01-10-M	These ACTIONS are changed to include the required ACTIONS for no required RHR loops OPERABLE. Previously, the specification did not provide specific action for no RHR loops OPERABLE. Also clarifies that the restored loop must be OPERABLE as well as in operation.	Yes	Yes	Yes	Yes
01-11-M	This change adds surveillances to verify breaker alignment and power availability to the required RHR pump not in operation.	Yes	Yes	Yes	Yes
01-12-M	The ACTIONS are changed to separate the required ACTIONS for only one required RHR loop OPERABLE and no required RHR loops OPERABLE.	Yes	Yes	Yes	Yes
01-13-M	An additional restriction is added which prohibits draining of the RCS when the RHR pump is deenergized.	No, see change 01-20-LS 27.	Yes	No, retaining CTS.	Yes
01-14-LS 22	This change modifies the LCO to only require two reactor coolant loops be OPERABLE in MODE 3. Required loops in operation depend on rod control system status.	No, two RCS loops OPERABLE is already part of the CTS.	No, two RCS loops OPERABLE is already part of the CTS.	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.4

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-15-M	A SG level corresponding to 10% of the wide range does not cover all SG tubes. To qualify as a valid heat sink, the tubes must be covered.	No, 15% water level is required for SG OPERABILITY in the CTS.	No, 10% narrow range S/G level is the CTS requirement.	Yes	Yes
01-16-A	This change revises the Note that permits up to 1 hour "deenergization" of RCP/RHR pumps. The revised wording clarifies the intent of the Note to allow the pumps to be "removed from operation" instead of "deenergized", thus permitting other means of removing the pumps from service.	Yes	Yes	Yes	Yes
01-17-LG	The plant specific minimum temperature specified in degrees below which the RCS must not be subject to overpressure is replaced by the generic statement "the minimum low temperature value specified in the PRESSURE TEMPERATURE LIMITS REPORT (PTLR)".	Yes	No, retaining CTS.	No, retaining CTS.	No, retaining CTS.
01-18	Not used	N/A	N/A	N/A	N/A
01-19-M	Changes allowed outage time for "four loops inoperable" from 72 hours to immediate action.	No, "immediate" already in CTS.	Yes	Yes	Yes
01-20-LS 27	The restriction for reactor vessel water level above the vessel flange to the stopping of all RHR pumps is removed.	Yes	No, CPSES does not have this restriction.	No, retaining CTS	No, Callaway does not have this restriction.

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-01-LS 3	The specification for the RCS safety valves in MODES 4 and 5 is deleted.	No, Amendments 98/97 relocated this specification to Equipment Control Guideline (ECG).	Yes	No, Amendment 89 relocated to USAR Chapter 16.	No, Amendment 103 relocated to FSAR Chapter 16.
02-02-M	This change extends the applicability for the required operability of the pressurizer safety valves to include MODE 4 when RCS temperature is above 320°F.	No, MODE 4 requirements are in the CTS.	Yes	No, WCGS has a different LTOP arming temperature.	No, Callaway has a different COMS arming temperature.
02-03-M	This change provides a new ACTION for two or more pressurizer safety valves inoperable. Prior to this change, LCO 3.0.3 would have been entered allowing for 1 hour prior to placing the unit in HOT STANDBY within the next 6 hours, etc.	Yes	Yes	Yes	Yes
02-04-LG	The footnote about the lift setting being for ambient conditions is moved to the Bases.	Yes	Yes	No, see 2-05-LS-23.	No, see 2-05-LS-23.
02-05-LS 23	Safety valve lift settings at ambient (hot conditions) can be deferred for 54 hours following MODE 3 entry.	No, ACTION b. in the CTS already provides 54 hours allowance.	Yes	Yes	Yes
03-01-LS 4	This change revises ACTION from the "pressurizer otherwise inoperable" to "water level not within limit".	Yes	Yes	Yes	Yes
03-02-LG	The method of verifying heater capacity is moved to the Bases.	Yes	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-03-LS 28	This change extends the pressurizer heaters surveillance interval from 92 days to 18 months.	Yes	Yes	Yes	No, already part of CTS per Amendment 105.
03-04-LS 29	The ACTION is modified to reflect generic wording to assure that the rods are fully inserted and cannot be withdrawn. This change is based on Industry Traveler TSTF-87, Rev. 1.	Yes	Yes	Yes	Yes
03-05-M	This change adds a requirement to CTS 3.4.3 LCO. The change requires the pressurizer heaters to be capable of being powered from an emergency power source.	Yes	No, pressurizer heaters normally aligned to emergency power.	No, pressurizer heaters normally aligned to emergency power.	No, pressurizer heaters normally aligned to emergency power.
04-01-LS 5	An allowance has been added that separate condition entry is allowed for each Power Operated Relief Valve (PORV).	Yes	Yes	Yes	Yes
04-02-LS 6	The allowance for PORV inoperability has been expanded from previously only allowing block valve isolation when the PORV was inoperable for seat leakage to allowing block valve isolation when the PORV is still capable of manually being cycled.	Yes	Yes	Yes	Yes
04-03-M	The times to reduce MODE is decreased by one hour.	Yes	Yes	Yes	Yes
04-04-LG	This change moves the requirement to perform channel calibration of PORV actuation instrumentation to a licensee controlled document.	Yes, moved to the FSAR.	Yes, moved to the TRM.	Yes, moved to the USAR.	Yes, moved to the FSAR.
04-05-LS 31	The shutdown requirement of CTS 3.4.4 would require the plant to reduce T_{avg} to <500°F within 12 hours, rather than go to MODE 4, to address the concern of entering [LTOP] LCO Applicability with inoperable PORVs. For consistency the shutdown requirements of CTS LCO [3.4.8] would be similarly revised.	Yes	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
04-06-LS 32	This change provides a 72 hour completion time to restore an inoperable block valve, with the PORV placed in manual control mode. The CTS requires the block valve to be restored within one hour or remove power from the PORV solenoid	Yes	No, already part of CTS.	No, already part of CTS.	No, already part of CTS.
04-07-LS 33	This change provides a two hour completion time for restoring an inoperable block valve when more than one block valve is inoperable, and 72 hours to restore the remaining valves. The CTS requires the inoperable block valves to be restored within one hour for one or more valves inoperable.	Yes	No, already part of CTS.	No, already part of CTS.	No, already part of CTS.
04-08-LS 34	Consistent with Traveler WOG-60, the requirement to perform the PORV and block valve cycling surveillances is revised such that the surveillances are only required to be performed in MODES 1 and 2.	Yes	Yes	Yes	Yes
4-09-LS-36	Consistent with Industry Traveler WOG-87, the requirement to perform the 92 day surveillance of the pressurizer block valves (i.e., perform one complete cycle of each block valve) is revised such that it is not required if the block valve is closed to meet ACTION a of the CTS LCO 3.4.4.	Yes	Yes	Yes	Yes
05-01-A	This change moves the Steam Generator Surveillances to SR 3.4.13.2 and the Administrative Controls Sections 5.5.9 and 5.6.10.	Yes	No, same as CPSES change 01-14-A for CTS Section 3/4.0.	Yes	Yes
05-02-A	LCO 3.4.5 is deleted. Steam Generator operability requirements in MODES 1-4 are specified in the RCS loop and leakage specifications.	Yes	No, CPSES does not have this specification.	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-03-A	The probe orientation is changed to state "tube end" in place of "point of entry" in the tube inspection definition under Acceptance Criteria.	Yes	No, same as CPSES change 01-15 for CTS Section 3/4.0.	Yes	Yes
06-01-M	This change adds the performance of an RCS water inventory balance every 24 hours as a new requirement when the [sump level detector] is inoperable.	Yes	Yes	Yes	Yes
06-02-LS 8	This change allows the performance of an RCS water inventory balance every 24 hours as an alternative to the requirement to perform 24 hour samples of the containment atmosphere when a required radioactivity monitor is inoperable.	Yes	Yes	Yes	Yes
06-03-A	This change adds the word "required" to clarify that only those detectors which are being used to satisfy the LCO must be demonstrated to be OPERABLE.	Yes	Yes	Yes	Yes
06-04-A	The word "Digital" has been deleted to be consistent with the terminology used in NUREG-1431 as it relates to CHANNEL OPERATIONAL TESTS.	No, "Digital" not included in CTS.	Yes	No, "Digital" not included in CTS.	No, "Digital" not included in CTS.
06-05-A	This change deletes the phrase "not isolated from the Reactor Coolant System" when referring to leakage through the SGs.	No, the phrase is not part of the CTS.	Yes	Yes	Yes
06-06-A	This change moves the LCO for CONTROLLED LEAKAGE (seal injection flow) from "Operational Leakage" to ITS LCO 3.5.5 "ECCS."	Yes	Yes	No, see CN 06-28-LG.	No, see CN 06-28-LG.
06-07-LG	This change moves the listing of RCS PIVs to a licensee controlled document.	Yes, moved to the FSAR.	Yes, moved to TRM	Yes, moved to USAR Chapter 16.	Yes, moved to FSAR Table 16.4-1.
06-08-LS 9	Adds clarification that the valves in the RHR flow path are not required to meet the PIV specification "when in, or during the transition to or from the RHR MODE of operation" in MODE 4.	Yes	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
06-09-LS 10	The LCO Applicability for CONTROLLED LEAKAGE (seal injection flow) is reduced to only MODES 1, 2, and 3 with the associated change in ACTION b.	Yes	Yes	No, see CN 06-28-LG	No, see CN 06-28-LG
06-10-LG	Discussion on how to perform the leak testing is moved to the Bases.	No, valve leak test discussion is not part of CTS.	Yes	Yes	Yes
06-11-LS 11	This change allows for the RCS flow path to be isolated by one valve within 4 hours and a second series valve within 72 hours.	Yes	Yes	Yes	Yes
06-12-M	This change adds a further restriction which requires that any valve(s), used to satisfy the action of isolating the low pressure system from the high pressure system when a RCS PIV is inoperable, must additionally be tested and meet the leakage criteria of the PIV.	Yes	Yes	Yes	Yes
06-13-LS 12	This change removes the specific surveillance requirement to monitor the RCS Leakage Detection System once per 12 hours.	Yes	Yes	Yes, Note - WCGS had 2 TS surveillances for rad monitor and containment sump.	Yes, Note - Callaway had 2 TS surveillances for rad monitor and containment sump.
06-14-A	This change moves the Surveillance for CONTROLLED LEAKAGE (seal injection flow) from "Operational Leakage" to "ECCS".	Yes	Yes	No, see CN 06-28-LG.	No, see CN 06-28-LG.
06-15-LS 13	This change removes the additional criteria that "no more than 96 hours shall elapse between any two successive inventory balances".	No. see CN 06-26-LS 30. DCPD has a different surveillance.	Yes	No, see CN 06-26-LS. WCGS has a different surveillance.	No, see CN 06-26-LS. Callaway has a different surveillance.

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
06-16-LS 14	This change removes the requirement for monitoring the reactor head flange leakoff system.	Yes	Yes	Yes	Yes
06-17-LG	The definition of steady state is moved to the Bases.	Yes	Yes	No, WCGS did not have this definition.	No, Callaway did not have this definition.
06-18-LS 15	This change relaxes the requirement for PIV testing following operation in MODE 5. The previous requirement was testing following 72 hours in MODE 5 which is revised to 7 days in MODE 5.	No, MODE 5 testing requirement is not part of CTS.	Yes	Yes	No, already in CTS per Amendment 105.
06-19-TR 3	This change removes the specific requirement for performing the PIV surveillance prior to returning a valve to service following maintenance, repair or replacement.	Yes	Yes	Yes	Yes
06-20-A	IST requirements are moved to Administrative Controls Section 5.5.8 of the improved ITS.	Yes	Yes	No, WCGS does not have this requirement.	No, Callaway does not have this requirement.
06-21 LS- 35	This change increases the RCP seal injection flow Completion Time from 4 to 72 hours, with a new added verification that at least 100% of the assumed charging flow remains available.	Yes	Yes	No, see CN 06-28-LG	No, see CN 06-28-LG
06-22-M	This change adds a new ACTION to isolate the affected RHR penetration within 4 hours if the RHR suction isolation valve interlock function is inoperable.	No, not part of current DCPPTS.	Yes	Yes	Yes
06-23-LS 25	The leakage detection system specification is revised such that the provisions of 3.0.4 are not applicable.	Yes	No, the non-applicability of 3.0.4 is already part of the CTS.	Yes	Yes
06-24-M	Revises ACTION to require going to COLD SHUTDOWN rather than HOT SHUTDOWN with an RCS pressure less than 600 psig.	No, the 600 psig ACTION is not part of the CTS.	No, the 600 psig ACTION is not part of the CTS.	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
06-25-LS 26	The Operation Leakage LCO has been modified to change allowed limit for RCS PIVs.	Yes	No, leakage limit of ≤ 5 gpm is already part of CTS.	Yes	No, already part of CTS per Amendment 66.
06-26-LS 30	The surveillance requirement for performing an RCS water inventory balance is modified to allow deferral of the water inventory balance such that it would be performed within 12 hours after achieving steady state conditions.	Yes	No, already part of the CPSES CTS.	Yes	Yes
06-27-A	RCS leakage detection system descriptions are revised for consistency with CTS LCO 3.3.3.1 and FSAR Sections 5.2.5.2.2 and 11.5.2.3.2.2.	No, current systems are applicable.	No, current systems are applicable.	Yes	Yes
06-28-LG	The CTS definition of CONTROLLED LEAKAGE would be deleted. RCP seal performance is moved to a licensee controlled document.	No, not in CTS.	No, not in CTS	Yes, moved to USAR.	Yes, moved to FSAR.
07-01-R	This change moves the RCS chemistry specification from the CTS to a licensee controlled document.	No, Amendments 98/97 moved the specification to ECG.	Yes, to be relocated to TRM.	No, Amendment 89 relocated to USAR Chapter 16.	No, Amendment 103 relocated to FSAR Chapter 16.
08-01-LS 16	This change revises the applicability of the specification to MODES 1, 2, or 3 with $(T_{avg}) \geq 500^{\circ}\text{F}$. The change deletes the requirement to perform an isotopic analysis for Iodine every 4 hours in MODES 4 and 5 and in MODE 3 below 500°F , whenever the reactor coolant exceeds its DOSE EQUIVALENT Iodine or Gross Specific Activity limits.	Yes	Yes	Yes	Yes
08-02-LS 17	This change adds an exception to LCO 3.0.4 when operating in ACTION A. This would allow MODE changes under conditions that the plant is anticipating a return to acceptable activity levels within the 48 hour Completion Time.	Yes	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
08-03-LS 18	This change revises the sample frequency from 72 hours to 7 days for performance of a gamma isotopic analysis.	Yes	Yes	Yes	Yes
08-04-M	This change revises the measurement of I-131, I-133 and I-135 to DOSE EQUIVALENT I-131.	Yes	Yes	Yes	Yes
08-05-LS 19	This change revises the performance of the surveillance of the specific activity of the RCS following a 15% power change to MODE 1 only.	Yes	Yes	Yes	Yes
08-06-A	The definition of E-bar is located in the Definition Section of NUREG-1431 and is not duplicated in this specification.	Yes	Yes	No, WCGS does not have this definition as part of Table 4.4-4.	No, Calloway does not have this definition as part of Table 4.4-4.
08-07-A	This change adds a note to clarify the time in which the sample is required to be taken upon meeting the minimum conditions to obtain the sample.	Yes	Yes	Yes	Yes
08-08-LG	The description of gross radioactivity analysis is moved to the Bases.	Yes	No, not in CTS	No, not in CTS.	No, ot in CTS.
09-01-LG	The TS figures which contain operational parameters related to the RCS P/T limits are moved to the RCS PTLR.	Yes	Yes	Yes	Yes
09-02-M	The ACTIONS for RCS pressure/temperature limits are broken into MODES 1, 2, 3, or 4 and at all other times. The MODES 1, 2, 3, or 4 ACTION is essentially the CTS. The ACTION which covers other times requires immediate action to restore the parameters and a determination of RCS acceptability prior to entering MODE 4.	Yes	Yes	Yes	Yes
09-03-M	The time allowed for performance of the engineering evaluation following exceeding the P/T limits has been defined as 72 hours.	Yes	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-04-LG	This change moves the 10 CFR 50 Appendix H, material surveillance program from the CTS to a licensee controlled document.	No, this paragraph is not in the CTS.	Yes, moved to PTLR.	Yes, Note: Table 4.4-5 was deleted by Amendment 57.	Yes, Note: Material surveillance program moved to FSAR Ch. 16. Table 4.4-5 was deleted by Amendment 76.
09-05-R	This change removes the pressurizer specification from the TS. The limits for the pressurizer are relocated to an licensee controlled document.	No, Amendments 98/97 relocated specification to ECG.	Yes, to be relocated to the TRM.	No, Amendment 89 relocated the specification to USAR Chapter 16.	No, Amendment 103 relocated the specification to FSAR Chapter 16.
09-06-M	A requirement is included to either isolate the accumulators or depressurize them to a pressure corresponding with the RCS temperature to preclude an RCS over pressure event.	Yes	Yes	Yes	Yes
09-07-TR 2	This change removes the requirement for a special report to be generated and submitted to the NRC following the mitigation of an RCS pressure transient.	Yes	Yes	Yes	Yes
09-08-LS 7	A Note was added to CTS 3.4.9.3 ACTION to allow two centrifugal charging pumps capable of injection into the RCS during pump swap operation for \leq 1 hour.	Yes, specific to DCPP	No, CTS allows 2 CCPs to be OPERABLE.	No, see change 09-17-LS.	No, see change 09-17-LS.
09-09-M	An additional ACTION is added when more than the allowable number of charging pumps are capable of injecting into the RCS including subsequent actions for RCS depressurization and venting if necessary.	Yes	Yes	No, see change 09-15-M.	No, see change 09-15-M.

TECHNICAL SPECIFICATION CHANGE			APPLICABILITY		
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-10-M	An additional ACTION is added for an accumulator which has not been isolated or depressurized which includes subsequent actions for RCS depressurization and venting if necessary.	Yes	Yes	Yes	Yes
09-11-M	An additional ACTION is added for the system inoperable for any other reason to include action for RCS depressurization and venting.	Yes	Yes	Yes	Yes
09-12-LS 20	An allowance has been included which provides for performance of the CHANNEL OPERATIONAL TEST on the PORV actuation channels within 12 hours after entering the applicability for the system. Previously this was required to be performed prior to entry.	Yes	Yes	Yes	Yes
09-13-A	This requirement has been moved to the IST program as required by the Administrative Controls Section of the ITS.	No, RHR suction valve testing is not part of the DCPD CTS.	Yes	Yes	Yes
09-14-M	An additional surveillance is added for verification of accumulator isolation.	Yes	Yes	Yes	Yes
09-15-M	An additional action and corresponding surveillance are added for one or more SI pumps or more than one centrifugal charging pump capable of injecting into the RCS and subsequent actions for RCS depressurization and venting if necessary.	Yes	No, different LTOP design.	Yes	Yes
09-16-M	Increases the frequency for verification of RCS conditions during heatup, cooldown and testing from once every hour to once every 30 minutes.	Yes	No, 30 minute surveillance interval already in CTS.	No, 30 minute surveillance interval already in CTS.	No, 30 minute surveillance interval already in CTS.

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
9-17 LS-24	Three Notes are added to LCO 3.4.9.3 to reflect CTS SR 4.5.3.2, LCO 3.5.4 ACTIONS and b, and the LCO 3.5.4 Applicability Note*. A fourth Note is also added to LCO 3.4.9.3 to correspond to the accumulator ACTION added under CN 09-10-M. The Note regarding CCP pump swap operations represents a relaxation since it would allow both CCPs to be capable of injecting into the RCS for up to 4 hours throughout [COMS] applicability.	No, see change 09-08-LS-7.	No, CTS allows 2 CCPs to be OPERABLE.	Yes	Yes
10-01-LS 21	The surveillance requirements associated with the RCS Structural Integrity specification are deleted.	No, Amendments 98/97 moved these surveillances to ECG.	Yes	No, Amendment 89 moved to USAR Chapter 16. Also see Section 6.8.5.b.	No, Amendment 103 moved to FSAR Chapter 16. Also see Section 6.8.5.b.
10-02-A	The RCP flywheel inspection requirement has been moved to Section 5.5.7 in the ITS.	No, Amendments 98/97 moved the RCP flywheel surveillances to CTS 6.8.4.i.	Yes	No, Amendment 89 moved to USAR Chapter 16 and CTS 6.8.5.b.	No, Amendment 103 moved to FSAR Chapter 16 and CTS 6.8.5.b.

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
10-03 LS-37	The Reactor Coolant Pump Flywheel Inspection Program is revised to provide an exception to the examination requirements in Regulatory Guide 1.14, Rev. 1. The exception (to Regulatory Position C.4.b(1) and C.4.b(2)) allows for an acceptable inspection method of either an ultrasonic volumetric, or surface examination. The inspection would be conducted at ten year intervals coinciding with the Inservice Inspection schedule required by ASME Section XI.	No, see CN 02-17-LS 1 in the ITS Section 5.0 package.	Yes	No, see CN 02-17-LS 1 in the ITS Section 5.0 package.	No, see CN 02-17-LS 1 in the ITS Section 5.0 package.
11-01-R	This change removes the RCS system vents specification from the CTS. The requirements for the RCS vents will be moved to an licensee controlled document.	No, Amendments 98/97 relocated requirement to Equipment Control Guidelines (ECG).	Yes, to be relocated to the TRM.	No, Amendment 89 relocated to USAR Chapter 16.	No, Amendment 103 relocated to FSAR Chapter 16.

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information may be moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3..." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R" (Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The LCO and associated surveillances have been revised to be less restrictive. Prior to the proposed change, the requirement for RCS loops OPERABLE was based on Reactor Trip Breaker (RTB) position. The intent of this requirement was to allow for less stringent operating criteria when rod withdrawal events were precluded via the RTBs being open. The proposed specification allows more freedom in how rod withdrawal is precluded and is thus less restrictive. However, the intent of using physical plant characteristics to prevent inadvertent rod withdrawal is not diminished. The specification now acknowledges that the typical rod control system can be effectively disabled by other means than opening the RTBs.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not alter the requirement to preclude rod withdrawal using physical plant characteristics. The specification does not allow administrative control or other means which could be conceived as less stringent. The specification does allow for alternative means to opening the RTBs for precluding rod withdrawal. These means, such as removing power to the rod control system, would preclude inadvertent rod withdrawal as effectively as opening the RTBs. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Inadvertent rod withdrawal accidents have been previously evaluated. This change does not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will preclude rod withdrawal with the same level of assurance that opening the RTBs provided. Therefore, this change does not involve a significant reduction in a margin of safety.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (Continued)

Based on the above safety evaluation, the activities associated with NSHC "LS1" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

Current TS 3.4.7, "MODE 5 Loops Filled", LCO has been modified by a Note which allows the removal of all RHR loops from operation during planned heatup to MODE 4 from MODE 5 when at least one RCS loop is in operation.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This proposed change adds an additional relaxation to allow the use of an operating RCS loop in lieu of an operating RHR loop in MODE 5 during planned heatup in preparation to proceed into MODE 4. The primary functions of RHR loop operation in MODE 5 are to remove decay heat and to prevent boron stratification. These functions can also be performed by an operating RCS loop which is a normal method of accomplishing the same functions when in MODE 4. In addition, at least one RHR loop must remain OPERABLE during the transition to MODE 4. This proposed change does not alter the ability of the RHR and the RCS systems to perform their function to remove heat and provide boron mixing in the RCS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accidents that are potentially associated with this proposed change, are those related to a loss of decay heat removal capability or to boron stratification. Since the use of an RCS loop in lieu of an RHR loop to perform the heat removal/boron stratification functions does not result in any reduction in performance or reliability, the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS2" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

ACTION b. is modified to specifically define "water level not within its limit" as the condition requiring the action instead of the "pressurizer otherwise inoperable." Thus for any other condition which could cause pressurizer inoperability, (except for inoperable heaters which is covered by ACTION a.) LCO 3.0.3 would apply. The net effect is to provide an additional hour (allowed under LCO 3.0.3 to prepare for an orderly shutdown) to correct any condition other than high pressurizer water level before performing the required plant shutdown. This change is acceptable because a high pressurizer water level is the primary condition in MODES 1, 2, and 3 which could lead to pressurizer inoperability and therefore requires the most immediate actions. High pressurizer level is indicative of a reduction in the size of the pressurizer bubble which could adversely effect RCS pressure transients. The actions required for high pressurizer level are unaffected by this change. Any other cause of inoperability is not likely to require immediate action and is adequately protected by LCO 3.0.3.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The pressurizer functions to control RCS pressure and to provide expansion volume for the RCS during normal operating transients and accident conditions. The OPERABILITY of the pressurizer is assumed in the initial conditions of all accident analyses.

The proposed change provides an additional hour to restore the pressurizer to operable (other than for high pressurizer water level) prior to taking required shutdown actions. The primary cause of pressurizer inoperability is high water level or loss of heaters. The actions required by these causes remains unchanged in the improved specifications. Other causes for inoperability are much less likely and adequate protection is provided by LCO 3.0.3. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accidents that are potentially associated with this proposed change, are those related to a loss of pressurizer function which could result in an RCS overpressurization event. This change does not introduce any new overpressure accidents and the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS 4" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The LCO has been modified to allow separate condition entry for each pressurizer power operated relief valve (PORV). This makes restoration times for each PORV consistent based on the specific condition of a PORV, as long as at least one PORV is capable of being manually cycled.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change adds a relaxation to the LCO by allowing separate condition entry for each PORV. No analysis credit is taken for automatic actuation of the PORVs in MODES 1, 2, or 3. Analysis credit is taken for manual operation of one PORV during the Steam Generator Tube Rupture (SGTR) accident. The specific relaxation allowed by this change could result in having a PORV inoperable and not capable of manual control for a maximum of 144 hours (not the same PORV, i.e., two PORVs becoming inoperable consecutively) as compared to a maximum of 72 hours if separate condition entry were not allowed. However, the capability to manually cycle at least one PORV in response to a SGTR will be maintained during the allowed time. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accidents that are potentially associated with this proposed change, are those related to a loss of pressurizer function which could result in an RCS overpressurization event. This change does not introduce any new overpressure accidents and the existing analyses remain valid. Thus, the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The manual actuation capability of one PORV is credited in the SGTR accident analyses for MODES 1, 2, or 3. At least one PORV will remain capable of being manually cycled during the allowed ACTION time. The margin of safety established by the LCO remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS 5" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6
10 CFR 50.92 EVALUATION FOR SPECIFIC
LESS RESTRICTIVE TECHNICAL CHANGE

The ACTIONS for an inoperable PORV as a result of excess seat leakage associated with the LCO has been changed to specifically address inoperability as a function of whether or not the valve is capable of being manually cycled. This relaxation of the conditions for ACTION entry associated with the PORVs only requires the operability of the manual actuation mode of the pressure reduction function of the PORV. The capability of the PORV to be manually cycled as a criteria in the ACTION requirements continues to maintain the ability of the valve to mitigate a Steam Generator Tube Rupture (SGTR) accident.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change adds a relaxation to the ACTIONS associated with the LCO by expanding the conditions under which an otherwise inoperable PORV may be used to satisfy SGTR accident analyses assumptions. The assumptions require the PORV to be capable of manual operation. No credit is taken for the automatic actuation of the PORV in MODES 1, 2, or 3. Credit is taken for manual operation of the PORVs during the SGTR accident. However, the capability to manually cycle the PORVs will be unaffected by this change. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accidents that are potentially associated with this proposed change, are those related to a loss of pressurizer function which could result in an RCS overpressurization event. This change does not introduce any new overpressure accidents and the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. Automatic actuation of the PORVs is not credited in the accident analyses for MODES 1, 2, or 3. The PORVs will remain capable of being manually cycled. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS 6" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The proposed change adds a relaxation to CTS 3.4.9.3 ACTION to allow two centrifugal charging pumps (CCP)s to be capable of injecting into the RCS during pump swap operation for ≤ 1 hour. The LTOP system assumes no more than one CCP capable of injection into the RCS. This change to allow two CCPs capable of injection provides smooth transition for switching operating pumps without interrupting charging flow to the RCS. This change, in accordance with NUREG-1431, and Industry Traveler WOG-51 is acceptable based on the close operator attention during this evolution and his understanding of the need to avoid RCS low temperature overpressure.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change allows the capability of two CCPs to inject into the RCS for up to 1 hour during pump swap operation. This allowance provides for pump switching while avoiding a RCS transient associated with stopping and then restarting the injection flow. This proposed change would not affect the initiating events assumed for accidents previously evaluated and would not affect the ability of plant equipment to perform its intended function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce new equipment, does not involve any physical alterations to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The results of analyzed events are not altered. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS 7" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8
10 CFR 50.92 EVALUATION FOR SPECIFIC
LESS RESTRICTIVE TECHNICAL CHANGE

This proposed change revises the ACTION requirements for LCO 3.4.5.1 to provide an alternative (RCS water inventory balance every 24 hours) to the requirement to perform 24 hour samples of containment atmosphere when a required radioactivity monitor is inoperable. The primary function of the leakage detection systems is to detect significant reactor coolant pressure boundary (RCPB) degradation as soon as practical. The detection initiates the investigation to determine the source and to limit the potential for further degradation. The TS requires multiple diverse systems to ensure that leakage from a variety of locations and leakage rates can be detected in sufficient time to take measures to place the plant in a safe condition. When a required radioactivity monitor is inoperable, the CTS provides compensatory action by requiring containment atmosphere samples be taken and analyzed every 24 hours to supplement indications from detection systems which are still operable [(e.g., sump level and containment cooler condensate flow)]. Another diverse means for determining RCS leakage is the RCS water inventory balance. When performed at the same frequency as the containment atmosphere grab sample, and in conjunction with other leak detection indications, this method provides an equal degree of confidence of the status of RCS leakage.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously?

The RCS water inventory balance is a passive method for determining RCS leakage within LCO limits and cannot initiate or increase the consequences of an accident. The daily RCS inventory balance provides as much assurance that leakage will be detected as the daily containment atmosphere samples provide and is only one of several detection indications available. The probability that RCPB leakage will be detected is not reduced by using the RCS water balance instead of the containment atmosphere samples. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accidents that are potentially associated with this proposed change, are those related to a loss of RCPB integrity. This change does not introduce any new mechanisms for causing such RCPB integrity loss or degrade protective measures which mitigate such accidents. The existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS 8" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The proposed change allows the delay of Pressure Isolation Valve (PIV) testing in the RHR system during entry into MODE 4 from MODE 5 until RHR operation is completed. The current surveillance allows an exception to required testing within 24 hours following flow through these PIVs, but also allowed exception to the provisions of Specification 4.0.4 for entry into MODES 3 and 4. Otherwise, testing would be required prior to entry into MODE 4. PIVs in the normal RHR flow paths used in MODES 4, 5, and 6 are not required until final RHR operation is terminated during plant heatup. Completion of RHR operation is considered to be one of the required prerequisite conditions necessary for the performance of check valve leakage testing.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change clarifies that the valves in the RHR flow path are not required to meet the PIV specification when in or during the transition to or from the RHR mode of operation in MODE 4. The same allowance was available in the original specification by an exception to Specification 4.0.4. In addition, using RHR through these specific flow paths is the required mode of operation during an accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC.LS9 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accidents that are potentially associated with this proposed change, are those related to a loss of RCS inventory. This change does not introduce any new accidents related to a reduction in RCS inventory and the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. RHR will remain capable of performing the safety function and the new requirement will continue to provide adequate assurance of that capability. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS 9" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The LCO 3.4.5.2 Applicability for CONTROLLED LEAKAGE (RCP seal injection flow) is reduced to only MODES 1, 2, and 3, where previously it had also applied in MODE 4. Associated ACTION b is modified to add requirements to reduce flow within limits within 4 hours or to be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during SI. The seal injection flow limit is not applicable for MODE 4 because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements. Therefore, RCP seal injection flow need only be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance. Since seal injection restrictions are not required in MODE 4, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accident that could potentially be affected by this proposed change is the LOCA. The LOCA assumptions are continue to be satisfied in MODES 1, 2 and 3 and the LOCA analyses results are not affected by this change. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS10" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS11
10 CFR 50.92 EVALUATION FOR SPECIFIC
LESS RESTRICTIVE TECHNICAL CHANGE

The revised ACTION c would require in the event of a Pressure Isolation Valve (PIV) leakage not within limits that the high pressure portion of the affected system be isolated from the low pressure portion in 4 hours using at least one valve. Then, within 72 hours, [the affected system must be isolated using a second valve or] the leaking PIV must be repaired. Otherwise, the unit must be in HOT STANDBY in 6 hours and COLD SHUTDOWN within the following 30 hours. This requirement is acceptable on the basis that isolation by a single valve for no more than 72 hours provides an acceptable level of safety during the 72 hour interval. The valve used to isolate the inoperable PIV must have been leak tested in accordance with CTS 4.4.6.2.2.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The valve used to isolate the inoperable PIV will be leak tested in accordance with the surveillance requirements. With the successful completion of this leak test requirement, there is sufficient assurance that a single valve can provide adequate isolation for the following 72 hours. [The requirement to employ a second series isolation valve within 72 hours restores the two valve isolation required by the CTS.] The interval during which only single valve isolation of high-to-low pressure isolation is provided is sufficiently short so as to not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accidents that are potentially associated with this proposed change are those related to a loss of RCS inventory. This change does not introduce any new accidents related to a reduction in RCS inventory and the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS11 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The requirement to use one valve within the first 4 hours followed by [a second valve] within 72 hours provides an acceptable level of safety. Using valves that have been verified to be within limits provides assurance that low pressure piping is protected from RCS pressure. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS 11" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS12
10 CFR 50.92 EVALUATION FOR SPECIFIC
LESS RESTRICTIVE TECHNICAL CHANGE

This change in conformance with NUREG-1431, removes the specific requirement to monitor the containment atmosphere particulate and gaseous radioactivity monitor and the containment structure sump inventory and discharge once per 12 hours from current Operational Leakage TS. However, improved SR 3.4.15.1 requires that a CHANNEL CHECK be performed on the containment radioactivity monitor channels on the same frequency so there is no change in requirements for those detectors. The [containment sump level detection system] is continuously monitored from the control room via available alarms and indicators. The condensate from the [containment coolers condensate collection system] can be measured to provide another indicator of RCS leakage.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change deletes the requirement to monitor the RCS Leakage Detection System once per 12 hours. Leak detection provides information that may indicate degradation of the RCS pressure boundary, however, leak detection cannot initiate any accident. The RCS Leakage Detection System is not credited in any safety analyses. The continued operation of the leakage detection function is assured by the diverse means of leakage detection that have been provided within the system and by the requirement that a RCS water inventory balance be performed every 72 hours. The original requirement to "monitor" the system did provide specific details about what constitutes monitoring. Any reduction in performance of the system by the removal of the "monitoring" requirement is not expected to be significant. Since information is available from diverse sources the removal of the surveillance does not negatively impact RCS leak detection. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-12 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accidents that could potentially be affected by this change are LOCA and MSLB. These accidents have previously been evaluated and remain valid. Thus, the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The leakage detection system is not credited in any accident analyses. The margin of safety established by the LCO remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS 12" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS 14 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The current requirement to monitor reactor head flange leakoff at least once per 24 hours would be deleted on the basis that leakage into this system is identified leakage to the reactor coolant drain tank. The RCS water inventory balance is the definitive surveillance for determining leakage as categorized by ITS LCO 3.4.13.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change removes the requirement to monitor reactor head flange leakoff at least once per 24 hours. This leakage is into the reactor coolant drain tank and is classified as identified leakage. Identified leakage is determined by performance of a RCS water inventory balance. The initial inventory balance is required within 12 hours following RCS steady state operation and every 72 hours thereafter. Flange leakoff does not provide an indicator of pressure boundary integrity and consequently, the removal of the monitoring does not involve the probability of loss of RCS water inventory. The flange leakage does not involve any initial assumptions used in the accident analyses. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accidents that are potentially associated with this proposed change, are those related to a loss of RCS inventory. This change does not introduce any new accidents related to a reduction in RCS inventory and the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. RCS leakage will continue to be adequately monitored via the RCS water inventory balance. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS14" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS 16
10 CFR 50.92 EVALUATION FOR SPECIFIC
LESS RESTRICTIVE TECHNICAL CHANGE

CTS [3.4.8.] "Specific Activity," Applicability would be revised to delete MODE 3 with RCS average temperature less than 500°F and MODES 4 and 5 (consistent with NUREG-1431). In addition, this change deletes the requirement to perform the once per 4 hour surveillance for DOSE EQUIVALENT I-131 in the event the gross specific activity limit is exceeded, in accordance with industry traveler TSTF-28. The latter is an unnecessary requirement since the ACTION requires the plant to exit the LCO's revised Applicability. This is acceptable because the offsite release of radioactivity in the event of a Steam Generator Tube Rupture (SGTR) is unlikely since the saturation pressure of the reactor coolant in MODE 3 with RCS temperature less than 500°F, and MODES 4, and 5 is below the lift pressure of the main steam safety and [SG atmospheric relief] valves. This is also consistent with existing ACTIONS a and b which require the unit to be placed in MODE 3 with T_{avg} less than 500°F in the event that specific activity cannot be restored to within limits in the required time.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change in applicable MODES will not affect any of the initiating events assumed for any of the accidents previously evaluated. The release of radioactivity in the event of a SGTR in MODES 3, 4 and 5 is unlikely since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety and [SG atmospheric dump] valves. The proposed change in applicable MODES for which this specification applies will not affect the ability of plant equipment to perform its intended function to mitigate the consequences of an accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce new equipment into the plant or alter the manner in which existing equipment will be operated. This change does not introduce new accidents and the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The release of radioactivity in the event of a SGTR in MODES 3 (below 500°F), 4 and 5 is unlikely since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety and [SG atmospheric dump] valves. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS16" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS17 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The proposed change revises CTS [3.4.8.] Specific Activity, by adding adds an exception to LCO 3.0.4 when operating within ACTION a. This exception would allow MODE changes under conditions when the plant is anticipating a return to acceptable activity levels within the allowed completion time of required by ACTION a.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change will modify the MODE change restrictions by excluding LCO 3.0.4 as applied to RCS specific activity and will not affect the initiating events assumed for the accidents previously evaluated. This exception is acceptable because of the low risk of an accident during the ACTION allowed time when specific activity may not be within limits. [] Adding the exception to LCO 3.0.4 will not affect the ability of plant equipment to perform its intended function to mitigate consequences of an accident. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce new equipment into the plant or alter the manner in which existing equipment will be operated. This change does not introduce new accidents and the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS17" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS18 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The proposed change revises the RCS sample frequency of a gamma isotopic analysis from 72 hours to 7 days. The gamma isotopic analysis measures the gross specific activity which includes both the degassed and gaseous gamma activities.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change from 72 hours to 7 days will not affect the initiating events assumed for the accidents previously evaluated. The seven day interval provides sufficient indication of adverse trends to allow appropriate action prior to exceeding the LCO during normal operating conditions. The 7 day frequency is acceptable based on the low probability of a gross fuel failure occurring that would significantly alter the analysis results. Changing the sample frequency from 72 hours to 7 days will not affect the ability of plant equipment to perform its intended function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce new equipment into the plant or alter the manner in which existing equipment will be operated. This change does not introduce new accidents and the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS18" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS19 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The proposed change would revise the MODES for which the Iodine analysis surveillance is required to be performed for the RCS following a 15% power change to MODE 1 only. Previously, the surveillance was also required when the plant was in MODES 2 or 3. A 15% power change for MODES 2 and 3 would only occur following a reactor trip from MODE 1. The required surveillance would then only be conducted following a reactor trip when normal spiking would be anticipated, yielding results that would not be indicative of the operational status of the RCS.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or
2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or
3. Involve a significant reduction in a margin of safety."

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change would revise the applicable MODES from 1, 2, and 3 to MODE 1 only. The RCS Iodine analysis surveillance is required to be performed following a 15% power change. The removal of the MODES 2 and 3 requirement for analyzing Iodine in the RCS will not affect the initiating events assumed for the accidents previously evaluated and will not affect the ability of plant equipment to perform its intended function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce new equipment into the plant or alter the manner in which existing equipment will be operated. This change does not introduce new accidents and the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS19
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS19" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS20
10 CFR 50.92 EVALUATION FOR SPECIFIC
LESS RESTRICTIVE TECHNICAL CHANGE

Current overpressure protection TS Surveillance [4.4.9.3.1.a] requires that a CHANNEL OPERATIONAL TEST (COT) be performed on the PORV actuation channel within 31 days prior to entering a condition in which the PORV is required to be operable. This Surveillance would be revised to be required within 12 hours after entering MODE 4 when the PORV is required OPERABLE and at least once per 31 days thereafter. Previously the COT was required to be performed prior to entry. The proposed change is consistent with NUREG-1431 and provides the benefit of not requiring the PORV actuation channel to be tested in MODES 1, 2, or 3 [] as would be required by the CTS.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change only affects the time when a COT has to be performed on the PORV actuation channels. The PORV COT is still required within 12 hours of entering the condition where the PORV LCO is applicable. A 12 hour allowance considers the unlikelihood of a low temperature overpressure event during this time. The time of performing the surveillance is not an input or assumption for the initiating events for the accidents previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce any new equipment into the plant or alter the manner in which existing equipment will be operated. This change does not introduce any new accidents and the existing analyses remain valid. Thus the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. In addition, performing the test under MODE 4 conditions when the PORV setpoint can be reduced is more appropriate than performing the test prior to MODE 4 conditions. The margin of safety established by the LCOs also remains unchanged. Therefore, this change does not involve a significant reduction in a margin of safety.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NHSC LS20
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS20" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS 25 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The current Leakage Detection System TS ACTION is revised such that the provisions of TS 3.0.4 are not applicable. This revision will allow entry into the Applicable MODES with only one of the Leakage Detection Systems OPERABLE, subject to the requirements of the ACTION statements. This change is consistent with NUREG-1431, and Industry Traveler TSTF-60, and is acceptable because of the diverse means to detect RCS leakage prescribed in the ACTION.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The primary function of the Leakage Detection System is to detect the presence of RCS leakage which may be indicative to degradation of the RCS pressure boundary (RCPB). The TS provides for multiple means of leak detection. Small magnitude leaks can be detected to provide time to determine the RCBP condition before possible progressive degradation can occur. Entry into the applicable MODES while subject to an ACTION statement will not have any effect on the status of the RCPB. The proposed change does not result in degradation in the performance of the leak detection function nor increase the number of challenges imposed on safety-related equipment assumed to function during an accident situation. The proposed change will not affect any event initiators nor will it affect the ability of any safety-related equipment to perform its intended function. The Leakage Detection Systems are passive and cannot initiate or increase the consequences of an accident. No credit is explicitly taken for these systems in the accident analyses.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change will not affect the ability of any safety-related equipment to perform its intended function. The proposed change does not introduce any new equipment, does not involve any physical alterations to any plant equipment, and does not involve any change in the method by which any safety-related system performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS25
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. No credit is explicitly taken for these systems in the accident analyses. The proposed change does not affect the assumptions used in the safety analyses. The values assumed for the analyses and therefore the results of the analyses remain valid.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS 25" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS 26 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

This LCO has been modified to change the allowed leakage limit for reactor coolant system (RCS) pressure isolation valves (PIVs). The LCO permits system operation in the presence of leakage through PIVs in amounts which do not compromise safety.

The RCS is isolated from other systems by valves. During plant life these interfaces can produce varying amounts of reactor coolant leakage through either normal operational wear or through mechanical deterioration. Increasing allowed leakage limits from up to 1 gpm to up to 5 gpm for all PIVs will not challenge the pressure relief capacity of interfacing systems. This amount of leakage is considered negligible when compared with the capacity of the pressure relief valves. PIV leakage limits apply to leakage rates for individual valves.

The basis for this LCO is the 1975 Reactor Safety Study which identifies potential intersystem loss of coolant accidents (LOCAs) as a significant contributor to the risk of core melt. A subsequent study evaluated various PIV configurations to determine the probability of intersystem LOCAs. This study concluded that periodic leak testing of the PIVs can substantially reduce intersystem LOCA probability.

The previous leakage limit of 1 gpm applied to all valve sizes, is considered arbitrary and is not an indicator of imminent accelerated deterioration or potential valve failure. A study concluded allowable leak rates based on valve size was superior to a single allowable value. The single value imposes an unjustified penalty on the larger valves without providing information on potential valve degradation. In addition, enforcing the single value criteria resulted in higher personnel radiation exposures because larger valves must be repaired in place.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This LCO has been modified to change the allowed leakage limit for RCS PIVs. The modification does not compromise the PIV integrity. The operability requirements of the PIVs and their ability to perform their intended functions are not changed and the change will not alter the probability of occurrence of an accident. The allowed leakage rates are not considered to be significant to the analyses and consequences of an accident.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS26 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not increase the possibility of an intersystem LOCA. The proposed change does not introduce any new equipment, does not involve any physical alterations to any plant equipment, and does not involve any change in the method by which any safety-related system performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The LCO permits system operation in the presence of leakage through PIVs in amounts which do not compromise safety. The proposed PIV leakage rate change does not affect the acceptance criteria for any analyzed event. The proposed change does not affect the assumptions used in the safety analyses. The values assumed for the analyses and therefore the results of the analyses remain valid.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS 26" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS27
10 CFR 50.92 EVALUATION FOR SPECIFIC
LESS RESTRICTIVE TECHNICAL CHANGE

The CTS LCO footnote for "COLD SHUTDOWN - Loops Not Filled" has been revised in accordance with NUREG-1431. Improved TS Note 1 replaces CTS footnote provision 3) that states a RHR pump may be de-energized for up to one hour provided the reactor vessel water level is above the reactor flange. The replacement states a RHR pump may be de-energized for up to one hour provided there are no RCS draining operations to further reduce the RCS water inventory. The two other provisions, 1) and 2), required to de-energize a RHR pump are not changed. CTS footnote, provision c., maintains vessel water inventory above the flange level to provide a heat sink while the pump is de-energized. This revised footnote is less restrictive because the water inventory would be less with the level lower than the vessel flange. However, condition 2) requires that the core outlet temperature is maintained at least 10°F below saturation temperature. The condition is also present in the ITS. Monitoring core outlet water temperature during de-energization of the RHR pump for up to one hour assures the water temperature will not reach a condition where the cooling characteristics may change.

An increase in RCS temperature would be anticipated due to decay heat when the RHR pumps are de-energized. NUREG-1431 LCO 3.4.8, Note 1, suspends any draining operation during the time the pumps are not operating, but are OPERABLE and specifies that the core outlet must be maintained at least 10°F below saturation temperature. These conditions maintain RCS inventory and temperature to assure adequate cooling. The RHR pumps are OPERABLE and can be put into operation quickly if needed.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change will not affect any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS27 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The proposed change does not introduce new equipment, does not involve any physical alterations to plant equipment, and does not involve any changes in the method by which any safety-related system performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the assumptions or the acceptance criteria for any analyzed event. The values used in the analyses and therefore the results of the analyses remain valid.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS 27" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS28
10 CFR 50.92 EVALUATION FOR SPECIFIC
LESS RESTRICTIVE TECHNICAL CHANGE

This change relaxes the surveillance requirement for verifying the capacity of the pressurizer backup heaters from 92 days to 18 months. This change is consistent with Generic Letter 93-05, "Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation" and Industry traveler TSTF-93, Rev. 1. This change is being proposed to reduce the testing burden and to minimize unnecessary operation of RCS pressure control equipment.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The operability of the pressurizer heaters enhance the capability of the plant to control RCS pressure and establish natural circulation. The purpose of the surveillance requirement is to detect potential pressurizer heater degradation. This is done by periodically demonstrating that pressurizer heaters are capable of producing power at their design rating, by testing the power supply output and by performing electrical checks on heater elements.

Heating elements are simple resistive devices which are not prone to complex failure modes. Moreover, heater banks are made up of a number of individually powered heater elements such that a common mode failure of the elements is unlikely. The normal power supply to the heaters is in use during normal power operations and a failure of the power supply would be immediately detectable independent of the heater surveillance. The low failure rate experience with pressurizer heater elements is indicative that the surveillance interval may be extended without loss in heater reliability. The change is consistent with [DCPP] operating experience. In addition, the frequency of capacity testing the pressurizer heaters is not an assumption or input to any of the initiating events for accidents previously evaluated. While the presence of a steam bubble in the pressurizer is an implicit initial assumption for many safety analyses, a natural circulation shutdown can be accomplished without the use of pressurizer heaters. Thus the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS28 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The method of plant operation is unaffected since the change only affects the frequency of testing. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS 28" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS29
10 CFR 50.92 EVALUATION FOR SPECIFIC
LESS RESTRICTIVE TECHNICAL CHANGE

The ACTION requirement has been revised to allow alternate methods for precluding rod withdrawal in lieu of requiring that the RTBs be open. The proposed change allows more flexibility in how rod withdrawal is precluded and is considered less restrictive. However, the intent of using physical plant characteristics to prevent rod withdrawal is not diminished. The specification now acknowledges that the rod control system can be effectively disabled by means other than opening the RTBs.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The specification does not allow administrative control or other means which could be conceived as less stringent. The specification does allow for alternative means to opening the RTBs for precluding rod withdrawal. These means if used would be as effective as opening the RTBs, such as removing power to the Rod Control System. Therefore there should be no increase in the probability or consequences of a previously evaluated accident.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

Inadvertent rod withdrawal accidents have been previously evaluated. This change does not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will preclude rod withdrawal with the same level of assurance that opening of the RTBs provided. No reduction in the margin of safety will result from this change.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS29" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS30 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

The function of the RCS water inventory balance is to monitor for unexplained RCS leakage that could be indicative of degradation in the RCS pressure boundary (RCPB). A RCS water inventory balance cannot be meaningfully performed unless the reactor is operating at steady state conditions. The surveillance for current leakage detection would be revised to allow deferring the RCS water inventory balance in the event of a transient until 12 hours after steady state conditions have been established.

Unplanned plant transients can make obtaining a meaningful inventory balance difficult. Other means of leak detection are available during the deferred water inventory balance to provide independent monitoring for RCS leakage. This change is less restrictive, but is practical and acceptable for safe operation and is consistent with Industry Traveler TSTF-116, Rev. 1.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change will not affect any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. A diversity of leak detection monitoring exists independent of the deferral of performing the water inventory balance. The water inventory balance does not initiate any safeguards actuation signals and is not involved with the operation of any safety-related equipment.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS30 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The water inventory balance is performed using input from existing RCS sensors. The proposed change does not introduce any new equipment, does not involve any physical alterations to any plant equipment, and does not involve any change in the method by which any safety-related system performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the assumptions or the acceptance criteria for any analyzed event. The values assumed for the analyses and therefore the results of the analyses remain valid.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS30" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS31 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

With inoperable PORVs (inoperable for the overpressure protection function of both CTS LCOs 3.4.4 and [3.4.9.3]), the requirement of LCO 3.4.4 to shutdown to MODE 4 will result in operation in the Applicability for LCO [3.4.9.3], which also requires the PORVs to be OPERABLE. It is undesirable, from a safety perspective, to intentionally enter the LCO [3.4.9.3] Applicability with inoperable overpressure protection. Therefore, a change is proposed that would only require the shutdown to MODE 3 with RCS $T_{avg} < 500^{\circ}\text{F}$ within 12 hours, where the function of the PORVs to mitigate an SGTR is no longer needed. The proposed change would also include a continuing requirement to undertake actions to restore inoperable valve(s). The risks of entering the LCO [3.4.9.3] Applicability with inoperable PORVs is greater than allowing plant shutdown to terminate at MODE 3 with $T_{avg} < 500^{\circ}\text{F}$. The offsite release of radioactivity in the event of an SGTR is unlikely with RCS $T_{avg} < 500^{\circ}\text{F}$ since the saturation pressure of the reactor coolant would be below the lift pressure of the main steam safety and [SG atmospheric dump] valves. The shutdown requirement for CTS [3.4.8] would similarly be revised. This change is consistent with Industry Traveler TSTF-113.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility, involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the ACTIONS associated with CTS LCO 3.4.4 by keeping the end point of the shutdown action above the LCO [3.4.9.3] Applicability. No credit is taken for the automatic actuation of the PORVs in MODES 1, 2, or 3. Credit is taken for the manual operation of the PORVs during a SGTR. The capability to manually cycle the PORVs will be unaffected by this change. The release of radioactivity in the event of an SGTR with RCS $T_{avg} < 500^{\circ}\text{F}$ is unlikely since the saturation pressure of the reactor coolant would be less than the lift pressure of the main steam safety and [SG atmospheric dump] valves. Since the Bases for CTS [3.4.8] are also associated with the SGTR, shutdown requirements would be revised to require MODE 3 be reached within 6 hours and T_{avg} reduced to $< 500^{\circ}\text{F}$ within 12 hours. The proposed change to the ACTION termination point of CTS 3.4.4 and the increased Completion Time of CTS [3.4.8] will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. The proposed change in the ACTION statement will not affect any of the analysis assumptions for any of the accidents previously evaluated. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS31 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in ACTION termination point will present no undue burden on the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The release of radioactivity in the event of an SGTR with RCS $T_{avg} < 500^{\circ}\text{F}$ is unlikely since the saturation pressure of the reactor coolant would be less than the lift pressure of the main steam safety and [SG atmospheric dump] valves. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS31" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS32 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

CTS ACTION c for an inoperable block valve requires a inoperable PORV block valve to be restored within one hour or close the PORV and remove power from its solenoid. This change, in accordance with NUREG-1431, requires the PORV to be placed in manual control mode and provides a 72 hour completion time to restore an inoperable block valve.

The action to close the PORV and remove power from its solenoid loses the function of the PORV as it can not be cycled in either the automatic or in the manual control mode. CTS ACTION for an inoperable PORV requires the PORV to be restored in 72 hours. Removing the power from the PORV solenoid makes the valve inoperable and consequently the valve would enter the above ACTION statement. Although the 72 hour limit is provided to restore the PORV, it also works for the block valve restoration. The PORV has been administratively disabled and remains so until the block valve is restored. Therefore the block valve must be restored within 72 hours to avoid shutdown under the PORV completion time.

Although the Completion Times can be seen to be equivalent, the ACTION taken on the PORV is different. The CTS removes the capability of the PORV. The valve can not open and there is no potential of a stuck open PORV without an OPERABLE block valve to isolate the PORV flow path. The ITS removes the automatic actuation capability to avoid the valve opening in the automatic actuation mode when the block valve is inoperable. The PORV however can still function in the manual mode to mitigate events such as the steam generator tube rupture. However, the valve being capable of being opened involves the potential for failure to close. Consequently, this change is considered to be less restrictive.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This proposed change provides 72 hours to restore an inoperable block valve with the associated PORV in the manual control mode. This time is considered to be reasonable based on the unlikelihood of a challenge to the PORVs during this time. Manual actuation of the PORV is unlikely due to the availability of the other PORVs. This change will not affect the initiating events assumed for accidents previously evaluated. This proposed change would not affect the ability of plant equipment to perform its intended function. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS32 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce new equipment, does not involve any physical alterations to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change provides for placing a PORV under manual control while a block valve is inoperable for up to 72 hours. Automatic valve actuation is not assumed in accident analyses and the analyses results are not affected. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, it is concluded that the activities associated with NSHC "LS32" resulting from the conversion of current TS 3/4.4 to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS33 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

CTS ACTION for one or more PORV block valves inoperable requires the inoperable block valves to be restored within 1 hour or close the associated PORVs and remove power from their solenoids. This change, in accordance with NUREG-1431, requires the associated PORVs to be placed in manual control mode and provides a 2 hour Completion Time to restore one inoperable block valve and a 72 hour Completion Time to restore all block valves.

The CTS requires block valve restoration within 1 hour or close the associated PORVs and remove power from their solenoids. This action loses the function of the affected PORVs as they can not be cycled in either the automatic or in the manual control mode. CTS ACTION for no Class I PORV OPERABLE requires at least one inoperable Class 1 PORV to be restored within 1 hour and the second one within 72 hours. Removing the power from the PORV solenoids makes the PORVs inoperable and consequently the Class 1 valves would enter the appropriate ACTION as a consequence of the required ACTION for an inoperable block valve. The PORVs would be administratively disabled and if not restored within 1 hour, plant shutdown would be initiated.

The CTS removes the ability of the PORV to function. The valve cannot open and there is no potential of a stuck open PORV without an OPERABLE block valve to isolate the PORV flow path. However, the improved TS removes only the automatic actuation capability to avoid the valve opening in that mode when the block valve is inoperable. The PORV however can still function in the manual mode to support SGTR mitigation. The valve being capable of manual operation involves the potential for failure to close. A failure to close can not be isolated due to the associated block valve being inoperable. The time for which this condition exists is 2 hours in accordance with the ITS whereas the CTS requires block valve restoration in 1 hour. Consequently, this change is considered to be less restrictive.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This proposed change provides 2 hours to restore one inoperable block valve with the associated PORV in the manual control mode. During this time the block valve cannot function to isolate the associated PORV if it were actuated manually and failed to close. This time is considered to be reasonable based on the unlikelihood of a challenge to the PORVs during this time. This change will not affect the initiating events assumed for accidents previously evaluated. This proposed change would not affect the ability of plant equipment to perform its intended function. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS33 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce new equipment, does not involve any physical alterations to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change provides for placing PORVs under manual control in the event of inoperable block valves. Automatic valve actuation is not assumed in accident analyses and the analyses results are not affected. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS 33" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS34 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

Consistent with Industry Traveler WOG-60, the requirement to perform the 92 day surveillance of the pressurizer PORV block valves and the 18 month surveillance of the pressurizer PORVs (i.e., perform one complete cycle of each valve) is revised to indicate that the surveillance is only required to be performed in MODES 1 and 2. This allows entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODES 1 or 2. This is consistent with the recommendations of Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10CFR50.54(f)," June 25, 1990., which indicates that administrative controls require this test be performed in MODES 3 or 4 to adequately simulate operating temperature and pressure effects on valve operation.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The Pressurizer PORV LCO requires the PORVs and block valves to be OPERABLE for manual operation to mitigate the effects associated with a SGTR. The SGTR is primarily a concern during MODES 1 and 2 because the reactor is producing heat, transients are more likely, and the consequences of an accident are more severe. In MODE 3, due to the stable conditions of the plant, a SGTR event is considered to be highly unlikely. Also, the number of PORVs required for heat removal to mitigate a SGTR event is less in MODE 3. More importantly, this surveillance assures that the PORV will operate when called upon under the design basis conditions. These conditions are more appropriately simulated in MODE 3. While the probability of a SGTR event is not affected by this change the consequences may increase slightly if a SGTR occurred in MODE 3 and one or more untested PORVs failed to operate. The slight increase in consequences is not considered significant. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS34
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. By allowing testing in MODE 3, the proposed change provides additional assurance that the PORVs and block valves will be OPERABLE when called upon by allowing testing closer to the design basis conditions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS 34" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS35 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

This change increases the seal injection flow Completion Time from 4 to 72 hours, with a new added verification that at least 100% of the assumed charging flow remains available. The Bases for seal injection flow relate the limit to ensuring adequate charging flow during post-LOCA injection. The revised ACTIONS continue to assure this basis is adequately addressed by providing an ECCS-like Required Action. Specification 3.5.2 allows a 72 hour Completion Time for one or more ECCS subsystems inoperable if at least 100% of the assumed ECCS flow is available. The seal injection flow ACTIONS have been modified so that if the remaining charging flow (with some inoperability in the charging system) is greater than or equal to 100% of the assumed post-LOCA charging flow, 72 hours is allowed to restore OPERABILITY. This change is consistent with Industry Traveler WOG-84.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change revises the completion time for restoring seal injection flow from 4 hours to 72 hours. The basis of this completion time is to ensure availability of the assumed post-LOCA charging flow. To compensate for the increased completion time, a new requirement is added to verify, within 4 hours, that at least 100% of the assumed post-LOCA charging flow is available. Since the change continues to ensure 100% of the assumed charging flow is available, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. Since the change continues to ensure 100% of the assumed charging flow is available, no new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS35
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. Since the change continues to ensure 100% of the assumed charging flow is available there will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS 35" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS36 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

Consistent with Industry Traveler WOG-87, the requirement to perform the 92 day surveillance of pressurizer PORV block valves (i.e., perform one complete cycle of each block valve) is revised such that it is not required if the block valve is closed to meet ACTION a of the CTS LCO 3.4.4.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change adds a relaxation to the surveillance associated with the pressurizer PORV block valves. The quarterly valve cycling will no longer be required if the block valve is closed per any ACTION of the LCO. No credit is taken for the automatic actuation of the PORV in MODES 1, 2, or 3. Credit is taken for manual operation of the PORVs during the SGTR accident. However, the capability to manually cycle the PORVs will be unaffected by this change. This change will not affect the ability of the block valve to open, if closed to meet ACTION a, in the mitigation of an SGTR. Deferral of the block valve cycling surveillance will not diminish the design capability of the block valve to open against differential pressures that would be present after an SGTR since the block valves are capable of opening against 2485 psig, the safety valve lift pressure, whereas pressurizer pressure decreases after an SGTR []. The lack of quarterly block valve cycling, which could extend to a complete cycle since ACTION a allows continued operation with the block valves closed, does not decrease the likelihood of successful pressurizer relief since power remains available to the block valve motor operator(s) and the surveillance frequency for the PORVs can be as long as 18 months (tested during each cold shutdown per the IST plan). Quarterly cycling could make PORV seat leakage worse. If the block valve were to subsequently be unable to close, this surveillance could unnecessarily challenge RCS and Pressurizer Relief Tank (PRT) integrity. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The only accidents that are potentially associated with this proposed change are those related to a loss of pressurizer relief function. This change does not introduce any new overpressure accidents and the existing analyses remain valid. Thus, the proposed change does not create the possibility of a new or different kind of accident from those previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS36
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The automatic actuation of the PORVs is not credited in the accident analyses for MODES 1, 2, or 3. The PORVs will remain capable of being manually cycled. The margin of safety established by the LCOs also remains unchanged. Thus there is no reduction in the margin of safety from that previously established.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS 36" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

V. GENERIC TECHNICAL NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR2 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

This change in accordance with NUREG-1431 removes the requirement for a special report to be generated and submitted to the NRC following the mitigation of a RCS pressure transient. Reporting to the NRC will be done commensurate with the reporting requirements of 10 CFR 50.72 and 50.73.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change is purely an administrative reporting change and cannot affect any accident probability or consequences. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

This change is purely an administrative reporting change and cannot create any new accident or affect any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change is purely an administrative reporting change. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "TR2" resulting from the conversion of current TS 3/4.4 to the improved TS format are concluded to satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

V. GENERIC TECHNICAL NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR3 10 CFR 50.92 EVALUATION FOR SPECIFIC LESS RESTRICTIVE TECHNICAL CHANGE

This proposed revision is to remove reference to specific post-maintenance tests from the CTS. Post-maintenance testing programs are controlled via plant administrative procedures in accordance with Licensee Controlled Document (ITS Section 5.4.1, Final Safety Analysis Report and Operating Quality Assurance Program) commitments to NRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)" and ANS 3.2-ANSI NIB.7. "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants". Specific post-maintenance testing requirements are contingent on the type and scope of maintenance actually performed as well as the availability and viability of test equipment, techniques, etc. Removal of specific testing requirements from the CTS and reliance on normal post-maintenance testing programs addressed by Licensee Controlled Documents allow flexibility to modify testing to address the circumstances of the maintenance performed while still assuring OPERABILITY of equipment returned to service,

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This is an administrative change which removes specific post-maintenance test requirements from the CTS. The testing, or equivalent testing, to assure equipment OPERABILITY prior to return to service would still be done as required by normal plant maintenance retest programs. Therefore, this change would not result in any increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

This is an administrative change and does not create a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change is an administrative change and does not affect any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "TR 3" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers (1 Page)

NUREG-1431 Specifications that are not applicable (1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
3.4.1	3.4-1 ¹
3.4.2	3.4-3 ²
3.4.3	3.4-5
3.4.4	3.4-7
3.4.5	3.4-8
3.4.6	3.4-11
3.4.7	3.4-14
3.4.8	3.4-17
3.4.9	3.4-19
3.4.10	3.4-21
3.4.11	3.4-23
3.4.12	3.4-27
3.4.13	3.4-33
3.4.14	3.4-35
3.4.15	3.4-39
3.4.16	3.4-43
3.4.17	N/A
3.4.18	N/A
3.4.19	N/A

Methodology (2 Pages)

Note (1): See Conversion for TS Section 3/4.2

Note (2): See Conversion for TS Section 3/4.1

Industry Travelers Applicable to Section 3.4

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-26	Incorporated	3.4-32	Approved by NRC
TSTF-27, Rev. 2	Incorporated	3.4-33	
TSTF-28	Incorporated	3.4-22	Approved by NRC
TSTF-54, Rev. 1	Not incorporated	N/A	
TSTF-60	Incorporated	3.4-15	Approved by NRC
TSTF-61	Not incorporated	N/A	Minor change that is adequately addressed in the Bases
TSTF-87, Rev. 1	Incorporated	3.4-31	
TSTF-93, Rev. 1	Incorporated	3.4-17	
TSTF-94	Not incorporated	N/A	Retained CTS
TSTF-105	Incorporated	3.4-38	
TSTF-108, Rev. 1	Not incorporated	N/A	LCO 3.4.19 does not apply
TSTF-113, Rev. 3	Incorporated	3.4-39	
TSTF-114	Incorporated	N/A	Approved by NRC
TSTF-116, Rev. 1	Incorporated	3.4-36	
TSTF-136	Incorporated	N/A	
TSTF-137	Incorporated	N/A	
TSTF-138	Not incorporated	N/A	Inconsistent with RCS loops requirements of ITS 3.4.5 and 3.4.6
TSTF-151	Incorporated	N/A	
TSTF-153	Incorporated	3.4-01	
TSTF-162	Incorporated	N/A	
WOG-51, Rev. 1	Incorporated	3.4-45, 3.4-23	See also CNs 3.4-18 and 3.4-20
WOG-60	Incorporated	3.4-35	
WOG-67, Rev. 1	Incorporated	3.4-10	DCPP only
WOG-87	Incorporated	3.4-47	
WOG-99	Incorporated	3.4-40	Applicable to Callaway and Wolf Creek only
WOG-100	Incorporated	3.4-49	

NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

Specification #	Specification Title	Comments
3.4.17	RCS Loop Isolation Valves	Plant design does not include RCS loop isoaltion valves.
3.4.18	RCS Isolated Loop Startup	Plant design does not include RCS loop isolation vlaves.
3.4.19	RCS Loops - Special Test Exceptions	This specification applies to tests that are only performed during initial plant startup and are no longer required.

RCS Pressure, Temperature, and Flow DNB Limits
3.4.1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

- a. Pressurizer pressure \geq ~~[2200]~~ 2197.3 psig; B-PS
- b. RCS average temperature \leq ~~[581]~~ 584.3°F; and B-PS
- c. RCS total flow rate \geq ~~[284,000]~~ gpm within limits shown on Table 3.4.1-1 for Unit 1 and Table 3.4.1-2 for Unit 2 3.4-1

APPLICABILITY: MODE 1.

-----NOTE-----
Pressurizer pressure limit does not apply during :

- a. THERMAL POWER ramp > 5% RTP per minute; or
 - b. THERMAL POWER step > 10% RTP.
-

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is \geq [2200] 2197.3 psig.	12 hours <u>B-PS</u>
SR 3.4.1.2	Verify RCS average temperature is \leq [581] 584.3°F.	12 hours <u>B-PS</u>

(continued)

RCS Pressure, Temperature, and Flow DNB Limits
3.4.1

SURVEILLANCE	FREQUENCY
SR 3.4.1.3, Verify RCS total flow rate is \geq [284,000] gpm within limits.	12 hours <u>3.4-41</u>
SR 3.4.1.4 <u>NOTE</u> Not required to be performed until 24 hours after \geq [90] % RTP. Verify by precision heat balance that measured RCS total flow rate is \geq [284,000] gpm within limits.	<u>3.4-51</u> [18] mont hs <u>B-PS</u> <u>3.4-38</u>

Table 3.4.1-1 (page 1 of 1)
Reduction in Percent RATED THERMAL POWER for Reduced RCS Flow Rate
Unit 1

3.4.1

RCS Total Flow ^(a) (10 ⁴ GPM)	Acceptable Operating Region ^(b) (% RTP)
≥ 35.9	≤ 100%
≥ 35.6	≤ 98%
≥ 35.2	≤ 96%
≥ 34.8	≤ 94%
≥ 34.5	≤ 92%
≥ 34.1	≤ 90%

(a) For RCS Total Flow < 341,000 GPM, entry into LCO 3.4.1 Condition A is required.

(b) When operating in the restricted power limits, the restricted power level shall be considered 100% RTP for Figure 2.1.1-1.

Table 3.4.1-2 (page 1 of 1)
Reduction in Percent RATED THERMAL POWER for Reduced RCS Flow Rate
Unit 2

3.4.1

RCS Total Flow ^(a) (10 ⁴ GPM)	Acceptable Operating Region ^(b) (% RTP)
≥ 36.3	≤ 100%
≥ 35.9	≤ 98%
≥ 35.5	≤ 96%
≥ 35.2	≤ 94%
≥ 34.8	≤ 92%
≥ 34.4	≤ 90%

(a) For RCS Total Flow < 344,000 GPM, entry into LCO 3.4.1 Condition A is required.

(b) When operating in the restricted power limits, the restricted power level shall be considered 100% RTP for Figure 2.1.1-1.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each ~~operating~~ RCS loop average temperature (T_{avg}) shall be \geq ~~541~~ $^{\circ}\text{F}$.

3.4-46

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

B

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more operating RCS loops not within limit.	A.1 Be in MODE 3 2 with $k_{eff} < 1.0$	30 minutes <u>3.4-32</u> <u>3.4-46</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} in each operating loop \geq 541 3 ³ °F.	<p style="text-align: center;"><u>B</u></p> <p style="text-align: center;"><u>3:4-46</u></p> <p style="text-align: center;">NOTE</p> <p style="text-align: center;">Only required if $T_{avg} = T_{ref}$ deviation, low low T_{avg} alarm not reset and any RCS loop T_{avg} \leq 547 °F</p> <p style="text-align: center;">-----</p> <p style="text-align: center;">12 hours - <u>3:4-33</u> 30 min utes thereafter</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Required Action A.2 shall be completed whenever this Condition is entered. ----- Requirements of LCO not met in MODE 1, 2, 3, or 4.</p>	<p>A.1 Restore parameter(s) to within limits. <u>AND</u> A.2 Determine RCS is acceptable for continued operation.</p>	<p>30 minutes 72 hours</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5 with RCS pressure \leftarrow [500] psig.</p>	<p>6 hours 36 hours <u>B</u></p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. -----NOTE----- Required Action C.2 shall be completed whenever this Condition is entered. ----- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.</p>	<p>C.1 Initiate action to restore parameter(s) to within limits. <u>AND</u> C.2 Determine RCS is acceptable for continued operation.</p>	<p>Immediately Prior to entering MODE 4</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.3.1 -----NOTE----- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ----- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.</p>	<p>30 minutes</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops—MODES 1 and 2

LCO 3.4.4 ~~Four~~ RCS loops shall be OPERABLE and in operation.

B

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify each RCS loop is in operation.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

LCO 3.4.5 ~~£ Two~~ RCS loops shall be OPERABLE, and either: B

- a. ~~£ Two~~ RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or B
- b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

-----NOTE-----
All reactor coolant pumps may be ~~de-energized~~ ~~removed from operation~~ 3.4.01
for \leq 1 hour per 8 hour period provided:

- a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
-

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. One required RCS loop not in operation, and reactor trip breakers closed and with Rod Control System capable of rod withdrawal.</p>	<p>C.1 Restore required RCS loop to operation.</p> <p><u>OR</u></p> <p>C.2 De-energize all control rod drive mechanisms (CRDMs). Place the Rod Control System in a condition incapable of rod withdrawal</p>	<p>1 hour <u>B</u></p> <p>1 hour <u>3.4.31</u></p>
<p>D. Two Four RCS loops inoperable.</p> <p><u>OR</u></p> <p>No RCS loop in operation.</p>	<p>D.1 De-energize all CRDMs. Place the Rod Control System in a condition incapable of rod withdrawal</p> <p><u>AND</u></p> <p>D.2 Suspend all operations involving a reduction of RCS boron concentration.</p> <p><u>AND</u></p> <p>D.3 Initiate action to restore one RCS loop to OPERABLE status and operation.</p>	<p>Immedia tely <u>B:PS</u> <u>3.4.31</u></p> <p>Immediately</p> <p>Immediately</p>

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	12 hours
SR 3.4.5.2	Verify steam generator secondary side water levels are \geq [17] 15% for required RCS loops.	12 hours <u>B:PS</u>
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

-----NOTES-----

1. All reactor coolant pumps (RCPs) and RHR pumps may be de-energized ~~removed from operation~~ for ≤ 1 hour per 8 hour period provided:
 - a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. No RCP shall be started with any RCS cold leg temperature \leq ~~[275]°F~~ the temperature below which LTOP is required as specified in the PTLR unless the pressurizer water level is less than 50% OR the secondary side water temperature of each steam generator (SG) is \leq ~~[50] °F~~ above each of the RCS cold leg temperatures.

3.4:01
3.4:10
3.4:19
ED

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable. AND Two RHR loops inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status. AND A.2 NOTE Only required if one RHR loop is OPERABLE Be in MODE 5.	Immediately 24 hours

3.4:02

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. One required RHR loop inoperable.</p> <p><u>AND</u></p> <p>Two required RCS loops inoperable.</p>	<p>B.1 Be in MODE 5.</p>	<p>24 hours <u>3:4:02</u></p>
<p>CB. Two required RCS or RHR loops inoperable.</p> <p><u>OR</u></p> <p>No RCS or RHR loop in operation.</p>	<p>G B.1 Suspend all operations involving a reduction of RCS boron concentration.</p> <p><u>AND</u></p> <p>G-B.2 Initiate action to restore one loop to OPERABLE status and operation.</p>	<p>Immedia tely <u>3:4:02</u></p> <p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.6.1 Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2 Verify SG secondary side water levels are \geq [17] 15% for required RCS loops.	12 hour s <u>B:PS</u>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.4.6.3 Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

- a. One additional RHR loop shall be OPERABLE; or
- b. The secondary side water level of at least ~~two~~ steam generators (SGs) shall be ~~≥ [17]~~ 15 %.

B
B:PS

-----NOTES-----

1. The RHR pump of the loop in operation may be ~~de-energized~~ removed from operation for ≤ 1 hour per 8 hour period provided:
 - a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.
3. No reactor coolant pump shall be started with ~~one or more~~ any RCS cold leg temperatures ≤ ~~[275]°F~~ the temperature below which LTOP is required as specified in the PTLR unless the pressurizer water level is less than 50% OR the secondary side water temperature of each SG is ≤ ~~≤ [50]~~ °F above each of the RCS cold leg temperatures.
4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

3:4:01

ED
3:4:10
3:4:19
B
ED

APPLICABILITY: MODE 5 with RCS loops filled

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. <u>AND</u> Required SGs secondary side water levels not within limits.	A.1 Initiate action to restore a second RHR loop to OPERABLE status.	Immediately
	<u>OR</u> A.2 Initiate action to restore required SG secondary side water levels to within limits.	Immediately
B. Required RHR loops inoperable. <u>OR</u> No RHR loop in operation.	B.1 Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	<u>AND</u> B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.7.1 Verify one RHR loop is in operation.	12 hours
SR 3.4.7.2 Verify SG secondary side water level is \geq [17] 15% in required SGs.	12 hours <u>B:PS</u>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.4.7.3 Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

-----NOTES-----

- 1. All RHR pumps may be de-energized ~~removed from operation for 15 minutes when switching from one loop to another~~ ≤ 1 hour provided:
 - a. ~~The core outlet temperature is maintained $> 10^{\circ}\text{F}$ at least 10°F below saturation temperature.~~ B
 - b. No operations are permitted that would cause a reduction of the RCS boron concentration; and
 - c. No draining operations to further reduce the RCS water volume are permitted.
- 2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

NOTE-

~~While this LCO is not met, entry into MODE 5, Loops Not Filled from MODE 5, Loops Filled is not permitted.~~

3.4.48

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required RHR loops inoperable.	B.1 Suspend all operations involving reduction in RCS boron concentration.	Immediately
<u>OR</u>	<u>AND</u>	
No RHR loop in operation.	B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.8.1 Verify one RHR loop is in operation.	12 hours
SR 3.4.8.2 Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LC0 3.4.9 The pressurizer shall be OPERABLE with:

- a. Pressurizer water level \leq ~~[92]~~ 90%; and B-PS
- b. Two groups of pressurizer heaters OPERABLE with the capacity of each group \geq ~~[125]~~ 150 kW ~~±~~ and capable of being powered from an emergency power supply. B-PS

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open.	6 hours	<u>3.4-31</u>
	<u>AND</u>		
	A.2 Fully insert all rods.	6 hours	
	<u>AND</u>		
	A.3 Place Rod Control System in a condition incapable of rod withdrawal.	6 hours	<u>3.4-31</u>
	<u>AND</u>		
	A.4 Be in MODE 4.	12 hours	
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.9.1 Verify pressurizer water level is \leq [92] <u>90%</u> .	12 hour s <u>B:PS</u>
SR 3.4.9.2 Verify capacity of each required group of pressurizer heaters is \geq [125] <u>150</u> kW.	92 days <u>18</u> months <u>B:PS</u> <u>3.4-17</u>
SR 3.4.9.3 [Verify required pressurizer heaters are capable of being powered from an emergency power supply]	18 months <u>B:PS</u> []

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 ~~Three~~ pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig. B

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 with all RCS cold leg temperatures $> 275^{\circ}\text{F}$. the temperature below which LTOP is required as specified in the PIR. 3.4:10

-----NOTE-----
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for ≤ 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup. B

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. <u>OR</u> Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with any RCS cold leg temperatures $\leq 275^{\circ}\text{F}$. the temperature below which LTOP is required as specified in the PIR.	6 hours 12 hours <u>3.4:10</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.10.1 Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within $\pm 1\%$.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY MODES 1, 2, and 3.

ACTIONS

- NOTES-----
1. Separate Condition entry is allowed for each PORV.
 2. LCO 3.0.4 is not applicable.
-

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour	
B. One or two PORV s inoperable and not capable of being manually cycled.	B.1 Close associated block valve s .	1 hour	<u>B-PS</u>
	<u>AND</u>		<u>B</u>
	B.2 Remove power from associated block valve s .	1 hour	<u>B</u>
	<u>AND</u>		<u>3.4.21</u>
	B.3 Restore the Class I PORV s to OPERABLE status.	72 hours	<u>B-PS</u>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. One block valve inoperable.</p>	<p>C.1 Place associated PORV in manual control.</p>	<p>1 hour</p>
	<p><u>AND</u></p> <p>C.2 If the block valve is associated with a Class I PORV. Restore block valve to OPERABLE status.</p> <p><u>OR</u></p>	<p>72 hours</p> <p style="text-align: right;"><u>3.4-21</u></p>
	<p>C.3 If the block valve is associated with the non-Class I PORV. Close the block valve and remove its power.</p>	<p>72 hours</p> <p style="text-align: right;"><u>3.4-21</u></p>
<p>D. Required Action and associated Completion Time of Condition A, B, or C not met.</p>	<p>D.1 Be in MODE 3. Initiate action to restore Class I PORV and/or associated block valve(s) to OPERABLE status.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.3 Be in MODE 4. Reduce Tavg to < 500°F.</p>	<p>6 hours <u>Immediately</u> <u>3.4-21</u></p> <p>6 hours</p> <p>12 hours <u>3.4-39</u></p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. Two Class 1 PORVs inoperable and not capable of being manually cycled.</p>	<p>E.1 Close associated block valves. Initiate action to restore Class 1 PORVs to OPERABLE status.</p>	<p>1 hour <u>B-PS</u> Immediate <u>3:4-39</u> 1Y <u>3:4-21</u></p>
	<p>AND</p>	<p>1 hour <u>3:4-39</u></p>
	<p>E.2 Remove power from associated block valves. Close associated block valves</p>	<p>6 hours <u>1</u> hour <u>3:4-39</u></p>
	<p>AND</p>	<p>12 hours <u>6 hours</u></p>
	<p>E.3 Be in MODE 3. Remove power from associated block valves</p>	<p>12 hours <u>3:4-39</u></p>
	<p>AND</p>	<p>12 hours <u>3:4-39</u></p>
	<p>E.4 Be in MODE 3</p>	<p>12 hours <u>3:4-39</u></p>
	<p>AND</p>	<p>12 hours <u>3:4-39</u></p>
	<p>E.5 Reduce Tavg to <500°F</p>	<p>12 hours <u>3:4-39</u></p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>F. More than one block valve inoperable.</p>	<p>F.1 Place associated PORVs in manual control.</p>	<p>1 hour</p>
	<p><u>AND</u></p>	
	<p>F.2 Restore one block valve for a Class I PORV to OPERABLE status [if three block valves are inoperable].</p>	<p>2 hours <u>3:4:21</u> <u>B</u></p>
	<p><u>AND</u></p>	
<p>G. Required Action and associated Completion Time of Condition F not met.</p>	<p>F.3 Restore remaining block valve(s) for a Class I PORV to OPERABLE status.</p>	<p>72 hours <u>3:4:21</u></p>
	<p><u>OR</u></p>	
	<p>F.4 If the remaining block valve is associated with the non-Class I PORV, close the block valve and remove its power.</p>	<p>72 hours <u>3:4:21</u></p>
<p>G. Required Action and associated Completion Time of Condition F not met.</p>	<p>G.1 Be in MODE 3. Initiate action to restore block valve(s) to OPERABLE status.</p>	<p>6 hours Immediately <u>3:4:39</u></p>
	<p><u>AND</u></p>	
	<p>G.2 Be in MODE 3.</p>	<p>12 6 hours</p>
	<p><u>AND</u></p> <p>G.3 Reduce Tavg to < 500°F</p>	<p>12 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	<p>-----NOTE----- 1. Not required to be met performed with block valve closed in accordance with the Required Action of Condition A B or E or Required Actions C.3 and F.4. 2. Only required to be performed in MODES 1 and 2. ----- Perform a complete cycle of each block valve.</p>	<p>3:4:47 3:4:21 3:4:35 92 days</p>
SR 3.4.11.2	<p>-----NOTE----- Only required to be performed in MODES 1 and 2 Perform a complete cycle of each PORV.</p>	<p>3:4:35 [18] months <u>B</u></p>
SR 3.4.11.3	<p>Perform a complete cycle of each solenoid air control valve and check valve on the air accumulators in PORV control systems. Demonstrate OPERABILITY of the safety related nitrogen supply for the Class I PORVs.</p>	<p>[18] months B:PS 18 month <u>3:4:26</u> S</p>
SR 3.4.11.4	<p>NOT USED Verify PORVs and block valves are capable of being powered from emergency power sources.</p>	<p>[18] months B</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with ~~no safety injection pumps and a maximum of a maximum of [one] [high pressure injection (HPI)] pump [and one centrifugal charging pump] capable of injecting into the RCS and the accumulators isolated and one of the following pressure relief capabilities~~ either a or b below B:PS
3:4-49

- a. ~~Two RCS relief valves, as follows:~~
 - ~~1. Two Class I power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR,~~ 3:4-21
 - OR
 - ~~[2. Two residual heat removal (RHR) suction relief valves with setpoints > [436.5] psig and < [463.5] psig. or]~~ B:PS
 - ~~[3. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint > [436.5] psig and < [463.5] psig].~~ B:PS
- b. The RCS depressurized and an RCS vent of \geq [2.07] square inches. B:PS

-----NOTES-----

- 1. Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. 3:4-45
- 2. The limitation for a maximum of one charging pump capable of injecting into the RCS is not required for pump swap operation until 1 hour after completion of pump swap operation. 3:4-45

APPLICABILITY: ~~MODE 4 when all any RCS cold leg temperature are \leq [275]°F the temperature below which LTOP is required as specified in the PTLR~~ 3:4-07
~~MODE 5,~~ 3:4-10
~~MODE 6 when the reactor vessel head is on and the vessel head closure bolts are not fully de-tensioned.~~ 3:4-27

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Two One or more [HPI] safety injection pumps capable of injecting into the RCS.</p>	<p>A.1 Initiate action to verify a maximum of [one] [HPI] pump is zero safety injection pumps are capable of injecting into the RCS.</p>	<p>Immediately</p> <p><u>3:4:06</u> <u>B-PS</u></p>
<p>B. Two or more centrifugal charging pumps capable of injecting into the RCS.</p>	<p><u>NOTE</u> Two charging pumps may be capable of injecting into the RCS during pump swap operation for < 15 minutes.</p> <p>B.1 Initiate action to verify a maximum of [one] centrifugal charging pump is capable of injecting into the RCS.</p>	<p>Immediately</p> <p><u>3:4:45</u> <u>B</u> <u>B-PS</u> <u>3:4:06</u></p>
<p>C. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.</p>	<p>C.1 Isolate affected accumulator.</p>	<p>1 hour</p>

(continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. Required Action and associated Completion Time of Condition [C] not met.</p>	<p>D.1 Increase RCS cold leg temperature to >[275]°F. the temperature below which LTOP is required as specified in the PTLR.</p> <p>OR</p> <p>D.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.</p>	<p>12 hours</p> <p><u>3.4-10</u></p> <p><u>B-PS</u></p> <p>12 hours</p>
<p>E. One required RCS relief valve Class I PORV inoperable in MODE 4.</p>	<p>E.1 Restore required RCS relief valve Class I PORV to OPERABLE status.</p>	<p>7 days</p> <p><u>3.4-21</u></p>
<p>F. One required RCS relief valve Class I PORV inoperable in MODE 5 or 6 with the vessel head closure bolts not fully de-tensioned.</p>	<p>F.1 Restore required RCS relief valve Class I PORV to OPERABLE status.</p>	<p>24 hours</p> <p><u>3.4-21</u></p> <p><u>3.4-27</u></p>

(continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G. Two required RCS relief valves Class I PORVs inoperable.</p> <p><u>OR</u></p> <p>Required Action and associated Completion Time of Condition A, B, D, E, or F not met.</p> <p><u>OR</u></p> <p>LTOP System inoperable for any reason other than Condition A, B, C, D, E, or F.</p>	<p>G.1 Depressurize RCS and establish RCS vent of \geq [2.07] square inches.</p>	<p>8 hours</p> <p><u>B-PS</u></p> <p><u>3.4.21</u></p> <p><u>B</u></p> <p><u>B</u></p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.12.1 Verify a maximum of [one] [HPI] zero safety injection pumps is are capable of injecting into the RCS.</p>	<p>12 hours</p> <p><u>B-PS</u></p>
<p>SR 3.4.12.2 Verify a maximum of one centrifugal charging pump is capable of injecting into the RCS.</p>	<p>12 hours</p> <p><u>B-PS</u></p>
<p>SR 3.4.12.3 Verify each accumulator is isolated when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.</p>	<p>12 hours</p> <p><u>3.4.23</u></p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.12.4 NOT USED Verify RHR suction valve is open for each required RHR suction relief valve.</p>	<p>12 hours</p>
<p>SR 3.4.12.5 -----NOTE----- Only required to be performed when complying with LCO 3.4.12.b. ----- Verify required RCS vent \geq [207] square inches open.</p>	<p>12 hours for unlocked open vent valve(s). <u>3.4-49</u> <u>B</u> AND 31 days for vent valve(s) locked open, sealed or otherwise secured in the open position. <u>3.4-28</u></p>
<p>SR 3.4.12.6 Verify PORV block valve is open for each required Class I PORV.</p>	<p>72 hours <u>3.4-21</u></p>
<p>SR 3.4.12.7 NOT USED Verify associated RHR suction isolation valve is locked open with operator power removed for each required RHR suction relief valve.</p>	<p>31 days <u>B</u></p>
<p>SR 3.4.12.8. -----NOTE----- Not required to be met performed until 12 hours after decreasing any RCS cold leg temperature to \leq [275]°F the temperature below which LTOP is required as specified in the PTLR. ----- Perform a COT on each required Class I PORV, excluding actuation.</p>	<p><u>3.4-49</u> <u>B</u> <u>3.4-10</u> 31 days <u>3.4-21</u></p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.4.12.9 Perform CHANNEL CALIBRATION for each required Class I PORV actuation channel.	18 <u> </u> mont <u> </u> hs <u>34:21</u>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

- a. No pressure boundary LEAKAGE;
- b. 1 gpm unidentified LEAKAGE;
- c. 10 gpm identified LEAKAGE;
- d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and
- e. ~~500~~ gallons per day primary to secondary LEAKAGE through any B one SG.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.13.1 -----NOTE----- Not required to be performed in MODE 3 or 4 until 12 hours after establishment of steady state operation. -----</p> <p>Perform RCS water inventory balance.</p>	<p>NOTE Only requir ed to <u>3.4.36</u> be performed during steady state operation</p> <p>72 hours</p>
<p>SR 3.4.13.2 Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.</p>	<p>In accordance with the Steam Generator Tube Surveillance Program</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the residual heat removal (RHR) flow path when in,
or during the transition to or from, the RHR mode of operation.

ACTIONS

- NOTES-----
1. Separate Condition entry is allowed for each flow path.
 2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more flow paths with leakage from one or more RCS PIVs not within limit.</p>	<p>-----NOTE----- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.1 and be in the reactor coolant pressure boundary or the high pressure portion of the system.</p>	<p style="text-align: center;"><u>B</u></p> <p style="text-align: right;">(continued)</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.</p> <p><u>AND</u></p> <p>A.2.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.</p> <p><u>OR</u></p> <p>A.2.2 Restore RCS PIV to within limits.</p>	<p>4 hours</p> <p>72 hours <u>B</u></p> <p>72 hours <u>B</u></p>
B. Required Action and associated Completion Time for Condition A not met.	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>
C. RHR System autoclosure interlock function inoperable.	<p>C.1 Isolate the affected penetration by use of one closed manual or deactivated automatic valve.</p>	<p>4 hours <u>B:PS</u></p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.14.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. <p>-----</p> <p>Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure \geq 2215 psig and \leq 2255 psig.</p>	<p>In accordance with the Inservice Testing Program, and 18 months <u>B</u></p> <p><u>B</u></p> <p><u>AND</u></p> <p>Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months except for valves 8701 and 8702.</p> <p><u>AND</u></p> <p>(continued)</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.14.1 (continued)</p>	<p>Within 24 hours following valve actuation due to automatic or manual action or flow through the valve except for valves 8802A, 8802B and 8703</p> <p style="text-align: right;"><u>3.4.42</u></p>
<p>SR 3.4.14.2 NOT USED-----NOTE----- Not required to be met when the RHR System autoclosure interlock is disabled in accordance with SR 3.4.12.7.</p> <hr/> <p>Verify RHR System autoclosure interlock prevents the valves from being opened with a simulated or actual RCS pressure signal > [425] psig.</p>	<p style="text-align: center;"><u>B:PS</u></p> <p style="text-align: center;">[18] months</p>
<p>SR 3.4.14.3 NOT USED-----NOTE----- Not required to be met when the RHR System autoclosure interlock is disabled in accordance with SR 3.4.12.7.</p> <hr/> <p>Verify RHR System autoclosure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal > [600] psig.</p>	<p style="text-align: center;"><u>B:PS</u></p> <p style="text-align: center;">[18] months</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

- a. ~~One Both containment structure sumps and the reactor cavity sump level and flow monitor system. (level or discharge flow) monitor;~~ PS
- b. One containment atmosphere particulate radioactivity monitor (gaseous or particulate); ~~and~~ 3.4.14
- c. ~~Either a One containment air cooler condensate flow rate fan cooler unit (CFCU) condensate collection monitor or the containment atmosphere gaseous radioactivity monitor~~ PS

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

~~NOTE
LCO 3.0.4 is not applicable.~~ 3.4.15

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment sump monitor inoperable.	NOTE LCO 3.0.4 is not applicable.	<u>3.4.15</u>
	NOTE Not required until 12 hours after establishment of steady state operation.	<u>3.4.36</u>
	A.1 Perform SR 3.4.13.1.	Once per 24 hours
	<u>AND</u>	
	A.2 Restore required containment sump monitor to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Required containment atmosphere particulate radioactivity monitor inoperable.</p>	<p>----- NOTE ----- ECO 3.0.4 is not applicable ----- B.1.1 Analyze grab samples of the containment atmosphere. ----- <u>OR</u> B.1.2 NOTE Not required until 12 hours after establishment of steady state operation Perform SR 3.4.13.1. ----- <u>AND</u> B.2.1 Restore required containment atmosphere particulate radioactivity monitor to OPERABLE status. ----- <u>OR</u> B.2.2 Verify containment air cooler condensate flow rate monitor is OPERABLE. -----</p>	<p>3.4.15 B Once per 24 hours 3.4.14 3.4.36 Once per 24 hours 30 days B-PS 3.4.14 30 days B-PS</p>
<p>C. Required containment air cooler condensate flow rate monitor inoperable.</p>	<p>C.1 Perform SR 3.4.15.1. OR C.2 Perform SR 3.4.13.1.</p>	<p>Once per 8 hours Once per 24 hours B-PS</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
<p>CC Required containment atmosphere gaseous radioactivity monitor inoperable.</p> <p><u>AND</u></p> <p>Required containment air cooler condensate flow rate monitor inoperable.</p>	<p>CC.1.1 Analyze grab samples of the containment atmosphere</p> <p><u>OR</u></p> <p>CC.1.2 NOTE Not required until 12 hours after establishment of steady state operation</p> <p>Perform SR 3.4.13.1</p> <p><u>AND</u></p> <p>DC.2.1 Restore required containment atmosphere gaseous radioactivity monitor to OPERABLE status.</p> <p><u>OR</u></p> <p>DC.2.2 Restore required containment air cooler condensate flow rate collection monitor to OPERABLE status.</p>	<p>Once per 24 hours</p> <p>Once per 24 hours</p> <p>30 days</p> <p>Once per 24 hours</p> <p>30 days</p>	<p>3.4-14</p> <p>3.4-16</p> <p>3.4-36</p> <p>3.4-16</p> <p>3.4-14</p> <p>PS</p>
<p>ED Required Action and associated Completion Time not met.</p>	<p>ED.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>ED.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>	<p>3.4-14</p>
<p>FE All required monitors inoperable.</p>	<p>FE.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>	<p>3.4-14</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.15.1	Perform CHANNEL CHECK of the required containment atmosphere particulate and gaseous radioactivity monitors.	12 hours	<u>3.4.14</u>
SR 3.4.15.2	Perform GO CHANNEL FUNCTIONAL TEST of the required containment atmosphere particulate and gaseous radioactivity monitors.	92 days	<u>3.4.29</u> <u>3.4.14</u>
SR 3.4.15.3	Perform CHANNEL CALIBRATION of the required containment sump monitors.	[18] months	<u>B</u>
SR 3.4.15.4	Perform CHANNEL CALIBRATION of the required containment atmosphere particulate and gaseous radioactivity monitors.	[18] months	<u>B</u> <u>3.4.14</u>
SR 3.4.15.5	Perform CHANNEL CALIBRATION of the required containment air cooler condensate flow rate [CECU] condensate collection monitors.	[18] months	<u>B</u> <u>PS</u>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2.
MODE 3 with RCS average temperature (T_{avg}) \geq 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 specific activity > 1.0 μ Ci/gm.	----- LCO 3.0.4 is not applicable. -----	<u>ED</u>
	A.1 Verify DOSE EQUIVALENT I-131 specific activity within the acceptable region of Figure 3.4.16-1.	Once per 4 hours <u>ED</u>
	<u>AND</u> A.2 Restore DOSE EQUIVALENT I-131 specific activity to within limit.	48 hours <u>ED</u>
B. Gross specific activity of the reactor coolant not within limit. 100/E μ Ci/gm	B.1 Perform SR 3.4.16-2.	4 hour S <u>3:4:22</u>
	<u>AND</u>	<u>3:4:25</u>
	B.2 Be in MODE 3. with $T_{avg} < 500^\circ\text{F}.$	6 hour S <u>3:4:39</u>
	<u>AND</u> B.2 Reduce to $T_{avg} < 500^\circ\text{F}$	12 hours <u>3:4:39</u>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 specific activity in the unacceptable region of Figure 3.4.16-1.	C.1 Be in MODE 3. with $T_{avg} < 500^{\circ}F$. AND	6 hours <u>3:4:39</u>
	C.2 Reduce T_{avg} to $< 500^{\circ}F$	12 hours <u>3:4:39</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.16.1 Verify reactor coolant gross specific activity $\leq 100/E \mu Ci/gm$.	7 days
SR 3.4.16.2 -----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq 1.0 \mu Ci/gm$.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of $\geq 15\%$ RTP within a 1 hour period.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.4.16.3 -----NOTE----- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for \geq 48 hours. ----- Determine \bar{E} from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for \geq 48 hours.</p>	<p>184 days</p>

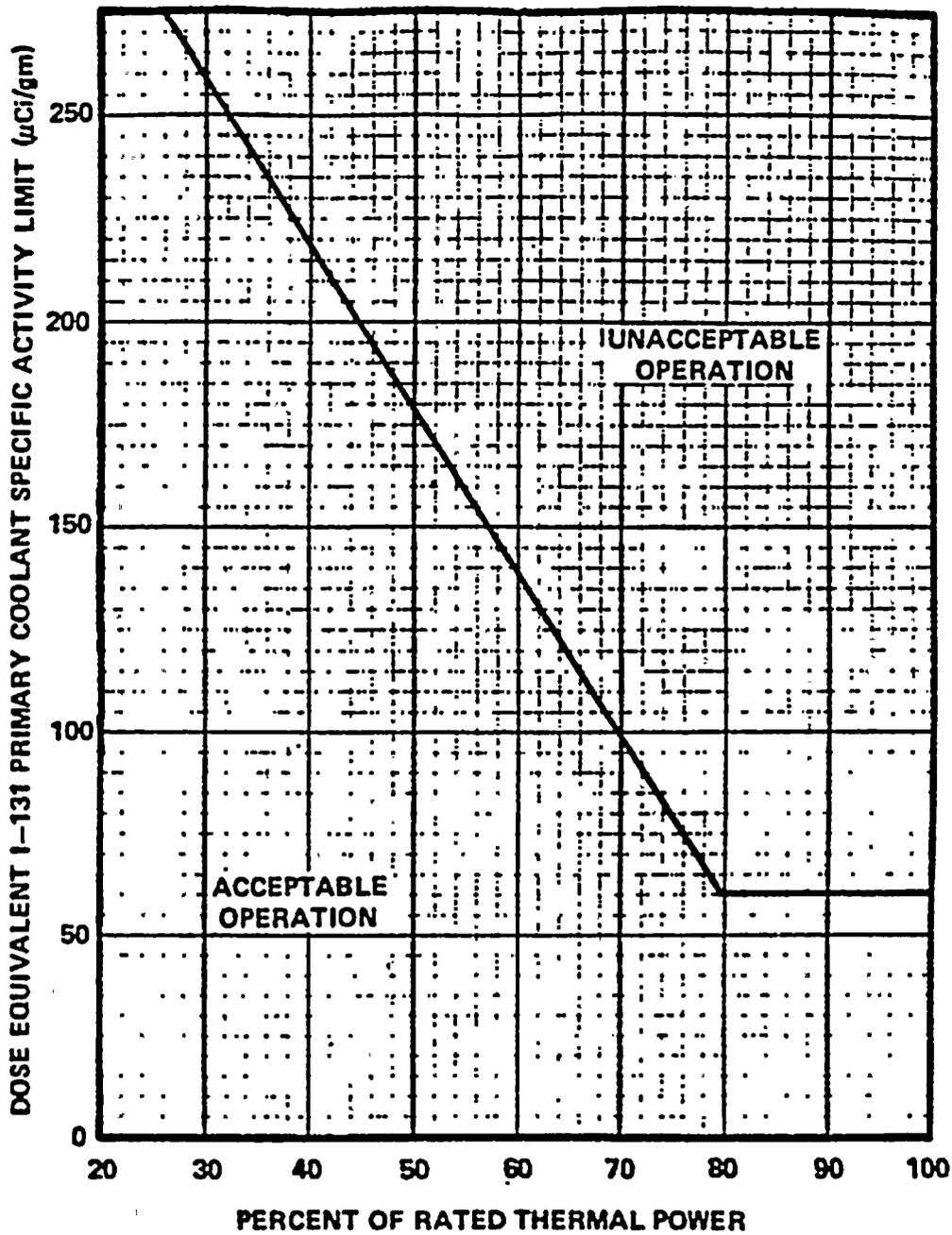


Figure 3.4.16-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT
VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT
SPECIFIC ACTIVITY $> 1 \mu\text{Ci/GRAM}$ DOSE EQUIVALENT I-131.

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes- Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

Methodology For Mark-up of NUREG-1431 Specifications
(Continued)

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES CONTENTS

Mark-up:

<u>BASES</u>	<u>PAGE</u>
3.4.1	B 3.4-1 ⁽¹⁾
3.4.2	B 3.4-6 ⁽²⁾
3.4.3	B 3.4-9
3.4.4	B 3.4-15
3.4.5	B 3.4-19
3.4.6	B 3.4-24
3.4.7	B 3.4-29
3.4.8	B 3.4-34
3.4.9	B 3.4-37
3.4.10	B 3.4-41
3.4.11	B 3.4-45
3.4.12	B 3.4-54
3.4.13	B 3.4-68
3.4.14	B 3.4-74
3.4.15	B 3.4-80
3.4.16	B 3.4-87
3.4.17	NA
3.4.18	NA
3.4.19	NA

Methodology

(2 Pages)

Note (1):See Conversion for TS Section 3/4.2

Note (2):See Conversion for TS Section 3/4.1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure
from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses and is variable with reactor thermal power down to 90% RTP as shown on Tables 3.4.1-1 and 3.4.1-2. Flow rate indications from the plant computer or RCS flow rate indicators are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits to be approached.

Operation for significant periods of time outside these DNB the limits on RCS flow, pressurizer pressure and average RCS temperature limits increases the likelihood of a fuel cladding failure if a DNB limited event were to occur.

APPLICABLE
SAFETY

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion correlation limit of $\geq [1.3]$ ≥ 1.17 (Ref. 2 and 3). This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criterion. The analyzed transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.7-6.

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSES
(continued)

"Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of [2200 2197.3] psig and the RCS average temperature limit of [581 584.3]°F correspond to nominal analytical limits of [2205] psig 2250 psia and [595] 577.6°F for Unit 2 (the limiting unit) used in the safety analyses, for the DNB calculation in the reload analyses with allowance for measurement analysis initial consideration uncertainty (38 psi and 6.7°F).

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement 10 CFR 50.36 (c) (2) (ii).

LCO

This LCO specifies limits on the monitored process variables--pressurizer pressure, RCS average temperature, and RCS total flow rate to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow limits are provided for a RTP range of 90% to 100% on Tables 3.4.1-1 and 3.4.1-2 for Unit 1 and Unit 2 respectively.

The RCS total flow rate limit allows for contains a measurement error of [2.34]%, based on performing a precision heat balance and using the result to normalize calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a non-conservative manner. Therefore, a bias error penalty of [0.1]%, for undetected fouling of the feedwater venturi raises the nominal flow is included in the measurement allowance to [2.34]%, for no fouling. error analysis

Any fouling that might significantly bias the flow rate measurement greater than [0.1]%, can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

(Continued)

BASES

LCO (continued) The LCO numerical values for pressure, temperature, and flow rate are given for the measurement location but have not been adjusted for instrument error.

APPLICABILITY In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational ~~pressure~~ transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and ~~reduce~~ the potential for violation of the accident analysis ~~limits~~.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal

(Continued)

BASES

ACTIONS
(continued)

condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition reduces the potential for violation of the accident analysis limits. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE
REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for the indicated RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions. The term "indicated RCS total flow" is used to distinguish between the "measured RCS total flow" determined in SR 3.4.1.4.

Sr 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance or other acceptable method once every [18] months allows the installed RCS flow instrumentation to be normalized and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

The Frequency of ~~[18]~~ months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance. Flow verification demonstrates that ~~setpoints are relevant and RCS flow resistance is within limits.~~

~~This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after > [90%] RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of [90%] RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching [90%] RTP.~~

REFERENCES

1. FSAR, Section ~~[15]~~.
 2. ~~Diablo Canyon Power Plant Unit 1 Cycle 9 Reload Safety Evaluation, August 1995~~
 3. ~~Diablo Canyon Power Plant Unit 2 Cycle 8 Reload Safety Evaluation, Rev. 1, April 1996~~
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is moderator temperature coefficient (MTC). LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

**APPLICABLE
SAFETY
ANALYSES**

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

All low power safety analyses assume initial RCS loop temperatures \geq the HZP temperature of 547°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

(Continued)

BASES

APPLICABLE SAFETY ANALYSES The RCS minimum temperature for criticality satisfies Criterion 2 of the NRC Policy Statement 10 CFR 50.36 (c) (2) (11).

(continued)

LCO Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a ~~with an operating loop~~ temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

The special test exception of LCO 3.1.8, "PHYSICS TESTS, Exceptions, MODE 2" permits PHYSICS TESTS to be performed at $\leq 5\%$ RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

ACTIONS A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE ~~3 2~~ with $k_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE ~~3 2~~ with $k_{eff} < 1.0$ in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS SR 3.4.2.1

~~RCS loop average temperature is required to be verified at or above [541]°F every 30 minutes when [T_{avg} - T_{cor} deviation, low low T_{avg}] alarm not reset and any RCS loop T_{avg} < [547]°F.~~

~~The Note modifies the SR. When any RCS loop average temperature is < [547]°F and the [T_{avg} - T_{cor} deviation, low low T_{avg}] alarm is alarming, RCS loop average temperatures could fall below the LCO requirement without additional warning. The SR to verify RCS loop average temperatures every 30 minutes is frequent enough to prevent the inadvertent violation of the LCO.~~

(Continued)

BASES

RCS loop average temperature is required to be verified at or above 541 °F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours is frequent enough to prevent inadvertent violation of the LCO and takes into account indications and alarms that are continuously available to the operator in the control room. If the $T_{avg} - T_{set}$ deviation were to alarm, the specific alarm response procedure would provide an increased frequency of monitoring. Following the clearance of the alarm, the frequency returns to 12 hours to monitor RCS T_{avg} .

REFERENCES 1. FSAR, Section [15.0.3] Chapter 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The ~~PRESSURE TEMPERATURE LIMITS REPORT~~ (PTLR) contains ~~pressure/temperature (P/T) limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.~~

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO ~~references the PTLR which~~ establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the ~~LCO-PTLR~~ limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials: ~~Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3). The NRC reviewed and approved methodology to be applied to determine P/T Limits is documented in the Administrative Controls Section 5.6.6.~~

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NOT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NOT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ~~ASTM E 195 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5) with the methodology identified in Section 5.6.6. The operating P/T limit curves will be adjusted, as necessary, based in agreement with the evaluation findings based and the recommendations of Regulatory Guide 1.99 (Ref. 6), on methods used in the PTLR.~~

(Continued)

BASES

BACKGROUND
(continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be $\geq 40^\circ\text{F}$ above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE
SAFETY
ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Administrative Controls Section 5.6.6 establishes identifies the NRC reviewed and approved methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of NRC Policy Statement 10 CFR 50.36(c) (2)(ii).

(Continued)

BASES

LCO

The two elements of this LCO are:

- a. The limit curves for heatup, cooldown, and ISLH testing; and
- b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

- a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;
- b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and
- c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and ~~Safety Limit 2.1~~, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

(Continued)

BASES

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 3), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

(Continued)

BASES

ACTIONS
(continued)

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. (Note that Action B.1 is not required when in MODE 4.) A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 73), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE
REQUIREMENTSSR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of

(Continued)

BASES

change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. ~~WCAP 14040 NP-A, Rev. 2, January 1996. Not Used~~
 2. 10 CFR 50, Appendix G.
 3. ASME, Boiler and Pressure Vessel Code, ~~Section III, Appendix G.~~ ~~Section XI, Appendix E~~
 4. ~~ASTM E 185-82, July 1982. NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Curves and Low Temperature Overpressure Protection System Limits" January 31, 1996.~~
 5. ~~10 CFR 50, Appendix H.~~
 6. ~~Regulatory Guide 1.99, Revision 2, May 1988.~~
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

- a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;
- b. Improving the neutron economy by acting as a reflector;
- c. Carrying the soluble neutron poison, boric acid;
- d. Providing a second barrier against fission product release to the environment; and
- e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The reactor coolant is circulated through ~~four~~ loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel ~~core~~ and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE
SAFETY
ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

~~Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming [four] RCS loops are in operation. The majority of the plant safety analyses are~~

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSES
(continued)

~~based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the [four] pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).~~

~~Steady state DNB analysis has been performed for the [four] RCS loop operation. For [four] RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 109% RTP. This is the design overpower condition for [four] RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 107% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.~~

~~All of the accident/safety analyses performed at RTP assume that all four RCS loops are in operation as an initial condition. Some accident/safety analyses have been performed at zero power conditions assuming only two RCS loops are in operation to conservatively bound lower modes of operation. The uncontrolled Rod Control Cluster Assembly (RCCA) Bank withdrawal from subcritical event is included in this category. While all accident/safety analyses performed at full rated power assume that all RCS loops are in operation, selected events examine the effects resulting from a loss of RCP operation. These include the complete and partial loss of forced RCS flow, RCP rotor seizure, and RCP shaft break events. For each of these events, it is demonstrated that all the applicable safety criteria are satisfied. For the remaining accident/safety analyses, operation of all four RCS loops during the transient up to the time of reactor trip is assured thereby ensuring that all the applicable acceptance criteria are satisfied. Those transients analyzed beyond the time of reactor trip were examined assuming that a loss of offsite power occurs which results in the RCPs coasting down.~~

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the SL ~~Safety Limit~~ value during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of NRC Policy Statement ~~10 CFR 50.36 (c) (2) (ii)~~.

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, ~~four~~ pumps are required at rated power.

(Continued)

BASES

LCO (continued) An OPERABLE RCS loop consists of an ~~one~~ OPERABLE RCP in operation providing forced flow for heat transport and an the associated SG, OPERABLE in accordance with the Steam Generator Tube Surveillance Program, with a water level within the limits specified in SR 3.4.5.2, except for operational transients. A RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops-MODE 3";
LCO 3.4.6, "RCS Loops-MODE 4";
LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE
REQUIREMENTSSR 3.4.4.1

This SR requires verification every 12 hours that each RCS loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

(Continued)

BASES

REFERENCES 1. FSAR, Section 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel core and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

APPLICABLE SAFETY ANALYSES

~~Whenever the reactor trip breakers (RTBs) are in the closed position and the Control Rod Drive Mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the Rod Control System. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM.~~

Therefore, in MODE 3 with RTBs in the closed position and the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least two RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSES
(continued)

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops-MODE 3 satisfy Criterion 3 of NRC Policy Statement ~~10 CFR 50.36 (c) (2) (ii)~~

LCO

The purpose of this LCO is to require that at least ~~two~~ RCS loops be OPERABLE. In MODE 3 with the RTBs in the closed position and Rod Control System capable of rod withdrawal, ~~two~~ RCS loops must be in operation. ~~Two~~ RCS loops are required to be in operation in MODE 3 with RTBs closed and the Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

~~With the RTBs in the open position, or the CRDMs de-energized, the Rod Control System is not capable of rod withdrawal; therefore, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that redundancy for heat removal is maintained. safety analyses limits are met.~~

The Note permits all RCPs to be ~~removed from operation~~ de-energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are ~~required to be performed without flow or pump noise, designed to validate various accident analyses values.~~ One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. ~~Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.~~

~~The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.~~

(Continued)

BASES

LCO (continued) Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial-startup test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with RTBs in the closed position, the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the RTBs open the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops-MODES 1 and 2";
- LCO 3.4.6, "RCS Loops-MODE 4";
- LCO 3.4.7, "RCS Loops-Mode 5, Loops Filled";
- LCO 3.4.8, "RCS Loops-Mode 5, Loops Not Filled";
- LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

BASES

ACTIONS
(continued)B.1

If restoration is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If the required RCS loop is not in operation and the Rod Control System is capable of rod withdrawal, the Required Action is either to restore the required RCS loop to operation or to place the Rod Control System in a condition incapable of rod withdrawal (e.g., de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets.) When the Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the Rod Control System must not be capable of rod withdrawal. The Completion Time of 1 hour to restore the required RCS loop to operation or to defeat the Rod Control System is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

D.1, D.2, and D.3

If four RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, place the Rod Control System in a condition incapable of rod withdrawal (e.g., all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets.) All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening

(Continued)

BASES

the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal.

ACTIONS
(continued) The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE
REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required loops are in operation. Verification may include flow rate, temperature, ~~or~~ and pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is $\geq 15\%$ for required RCS loops. If the SG secondary side narrow range water level is $< 15\%$, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

SR 3.4.5.3

Verification that the required RCPs are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.

REFERENCES None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops connected in parallel to the reactor vessel; each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel core and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. For RHR operation, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, a RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR loop circulates the water through the RCS at a sufficient rate to remove decay heat and to prevent boric acid stratification.

Although NUREG-1431 uses "loop" to define RHR system requirements, past practice is use of "train", consistent with ECCS discussions of train availability and redundancy. Plant procedures are written using "train". The designations of "loop" and "train" are considered synonymous.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS Loops - MODE 4 have been identified in NRC Policy Statement 10 CFR 50.36 (c) (2) (iii) as important contributors to risk reduction.

(Continued)

BASES

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be removed from operation de-energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that the secondary side water temperature of each SG be $\leq 50^{\circ}\text{F}$ above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature \leq the temperature below which LTOP is required as specified in the PTLR (the current limiting temperature for DCP is 270°F) 275°F . Note 2 also includes a DCP plant specific alternate condition under which a RCP may be started in MODE 4 and in MODE 5 with the loops filled. Note that RCPs may be "bumped" following a condition of RCS depressurization to establish "loops filled" condition. The Note specifies that a RCP may be started if the pressurizer water level is less than 50%. This option of RCP start with pressurizer water level less than 50% supports plant operational flexibility. The open volume in the pressurizer provides space to sustain reactor coolant thermal swell without incurring a possible excessive pressure transient due to energy additions from the S/G secondary water. The purpose of

(Continued)

BASES

LCO
(continued)

conditions to allow initial RCP start when none is running is to prevent a possible low temperature RCS overpressure event due to a thermal transient when a RCP is started. The condition of SG/RCS temperature difference limits the available relative energy source and the pressurizer level condition provides an expansion volume to accommodate possible reactor coolant thermal swell. ~~This~~ These restraint ~~is~~ conditions are intended to prevent a low temperature overpressure event due to a thermal transient when a RCP is started.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required. A RHR loop is in operation when the pump is operating and providing forced flow through the loop. Because a loop can be operating without being OPERABLE, the LCO requires at least one loop OPERABLE and in operation.

APPLICABILITY In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one required RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS loop or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(Continued)

BASES

If one required RHR loop is OPERABLE and in operation and there are no RCS loops OPERABLE, an inoperable RCS loop or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

If the parameters that are outside the limits cannot be restored, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 ($\leq 200^{\circ}\text{F}$) rather than MODE 4 ($>200^{\circ}\text{F}$ to $\leq 350^{\circ}\text{F}$). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

B.1 and B.2

If no loop is OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated.

~~Boron dilution requires forced RCS circulation from at least one RCP for proper mixing, and the margin to so that an inadvertent criticality must not be reduced in this type of operation may be prevented.~~ The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

**SURVEILLANCE
REQUIREMENTS****SR 3.4.6.1**

This SR requires verification every 12 hours that one RCS loop or RHR loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is $\geq 15\%$. If the SG secondary side narrow range water level is $< 15\%$, the tubes may become

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SG cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or two SGs with secondary side water levels above 15% to provide an alternate method for decay heat removal via natural circulation.

APPLICABLE
SAFETY
ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops - MODE 5 (Loops Filled) have been identified in NRC Policy Statement 10 CFR 50.36(c)(2)(ii) as important contributors to risk reduction

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level $\geq 15\%$. One RHR loop provides sufficient

(Continued)

BASES

LCO
(continued)

forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels $\geq 15\%$. Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation.

Note 1 permits all RHR pumps to be removed from operation de-energized ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise, designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be $\leq 50^{\circ}\text{F}$ above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with any an RCS cold leg temperature \leq the temperature below which Low Temperature Overpressure Protection (LTOP) is required as specified in the PTLR. Note that RCPs may be "bumped" following a condition of RCS depressurization to establish "loops filled" condition.

(Continued)

BASES

LCO
(continued)

Note 3 also includes an OR condition for starting a RCP. This condition is a DCCP plant specific alternate condition under which a RCP may be started in MODE 4 and in MODE 5 with the loops filled. The Note specifies that a RCP may be started if the pressurizer water level is less than 50%. This option of RCP start with pressurizer water level less than 50% supports plant operational flexibility. The open volume in the pressurizer provides space to sustain reactor coolant thermal swell without incurring a possible excessive pressure transient due to energy additions from the S/G secondary water. The purpose of conditions to allow initial RCP start when none is running is to prevent a possible low temperature RCS overpressure event due to a thermal transient when a RCP is started. The condition of SG/RCS temperature difference limits the available relative energy source and the pressurizer level condition provides an expansion volume to accommodate possible reactor coolant thermal swell. These conditions are intended to prevent a low temperature overpressure event due to a thermal transient when a RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be $\geq 15\%$.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";

LCO 3.4.5, "RCS Loops - MODE 3";

LCO 3.4.6, "RCS Loops - MODE 4";

LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";

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BASES

APPLICABILITY LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and

LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side water levels $< 15\%$, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Notes 1 and 4, Note 1, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent inadvertent criticality during a boron dilution, forced circulation from at least one RGP is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE
REQUIREMENTSSR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are $\geq 15\%$ ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

(Continued)

BASES

SURVEILLANCE
REQUIREMENT
(continued)

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side water level is $\geq 15\%$ in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None- 1 NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not filled) have been identified in the NRC Policy Statement ~~10 CFR 50.36 (c) (2) (11)~~ as important contributors to risk reduction.

LCO The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

Note 1 permits all RHR pumps to be de-energized for ~~removed from operation for < 1 hour 15 minutes when switching from one loop to another.~~ The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained at ~~least > 10°F below saturation temperature.~~ The Note prohibits boron dilution or draining operations when RHR forced flow is stopped.

(Continued)

BASES

LCO (continued) Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System. ~~The Applicability is modified by a Note stating that while the LCO is not met, entry into MODE 5, Loops Not Filled, from MODE 5, Loops Filled, is not permitted. This Note specifies an exception to LCO 3.0.4 and would prevent draining the RCS, which would eliminate the possibility of SG heat removal, while the RHR function was degraded~~

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODES 1 and 2";
 - LCO 3.4.5, "RCS Loops - MODE 3";
 - LCO 3.4.6, "RCS Loops - MODE 4";
 - LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
 - LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
 - LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).
-

ACTIONS

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required RHR loops are OPERABLE or in operation, except during conditions permitted by Note 1, all operations involving a reduction of RCS boron concentration must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. Boron dilution requires forced circulation from at least one RCP for uniform dilution, for proper mixing so that inadvertent ~~and the margin to criticality can be prevented. must not be~~

(Continued)

BASES

ACTIONS (continued) ~~reduced in this type of operation.~~ The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS SR 3.4.8.1

This SR requires verification every 12 hours that one loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.8.2

Verification that the required number of pumps are OPERABLE ensures that additional pumps can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls and emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. ~~Relatively small amounts of noncondensable gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.~~

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system.

Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

**APPLICABLE
SAFETY
ANALYSES**

In MODES 1, 2, and 3, the LCO requirement for a steam bubble reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensable gases normally present.

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSES
(continued)

Safety analyses presented in the FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of the NRC Policy Statement 10 CFR 50.36 (c) (2) (ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1600 [1240] cubic feet, which is equivalent to 90% of span ensures that a steam bubble exists. Instrument inaccuracy is not included in this % number. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 150 [125] kW, capable of being powered from either the offsite power source or the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. The capability to power the heaters from an emergency power supply using bus cross-tie to an OPERABLE emergency diesel generator, if necessary, provides the means to maintain system pressure control during a loss of normal power. RCS pressure control is necessary to maintain subcooling under conditions of natural circulation flow in the primary system. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops.

~~The exact design value of [125 kW is derived from the use of seven heaters rated at 17.9 kW each]. The amount needed to maintain pressure is dependent on the heat losses.~~

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply, and if

(Continued)

BASES

APPLICABILITY
(continued)

~~necessary, using bus cross-tie to an OPERABLE emergency diesel generator~~ In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1 and A.2, A.3 and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. ~~Normally the plant will trip in this event since~~ The upper limit of this LCO is ~~below the same as the~~ Pressurizer Water Level-High Trip at ~~92% of span~~.

If the pressurizer water level is not within the limit, action must be taken ~~to bring the unit to restore the plant to operation within the bounds of the safety analyses, a MODE in which the LCO does not apply.~~ To achieve this status, ~~within 6 hours the unit must be brought to MODE 3, with the reactor trip breakers open with rods fully inserted and the Rod Control System not capable of rod withdrawal, within 6 hours~~ Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the Applicable MODES, ~~and restores the unit to operation within the bounds of the safety analyses.~~

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station powered heaters.

ACTIONS
(continued)

C.1 and C.2

If one ~~required~~ group of pressurizer heaters ~~is~~ are inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating

(Continued)

BASES

ACTIONS (continued) experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within consistent with the safety analyses assumptions of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The Frequency of 92 days 18 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

SR 3.4.9.3

~~This SR is not applicable if the heaters are permanently powered by Class 1E power supplies.~~

~~This SR demonstrates that the heaters can be manually transferred from the normal to the emergency power supply and energized. The Frequency of 18 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.~~

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- REFERENCES
1. FSAR, Section 15.
 2. NUREG-0737, November 1980.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr at 2485 psig plus 3% accumulation [380,000] lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves which is divided equally among the three valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves and an increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, and 4 and 5. However, in MODE 4, with one or more RCS cold leg temperatures \leq the temperature below which LTOP is required as specified in the PTER (the current limiting temperature for DCPP is 270°F) [275]°F, and MODE 5 and MODE 6 with the reactor vessel head on and the reactor vessel head closure bolts not fully de-tensioned, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the $\pm 1\%$ of nominal pressure tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot (which is the current DCPP practice) or if valves are set cold that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include:

- a. Uncontrolled rod withdrawal from full power;
- b. ~~Feedline break~~ Loss of reactor coolant flow;
- c. Loss of external electrical load;
- d. Loss of normal feedwater;
- e. Loss of all AC power to station auxiliaries; and
- f. Locked ~~Reactor Coolant Pump (RCP)~~ rotor.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events b, c, d, e and f (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of ~~the NRC Policy Statement 10~~
~~CFR 50.36 (c)(2)(1)~~.

LCO

The ~~three~~ pressurizer safety valves are set to open at the RCS design pressure ~~(2485 psia) (2500 psia)~~, and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the $\pm 1\%$ of ~~nominal~~ pressure tolerance requirements (Ref. 1) for lifting pressures above 1000 psig.

The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of ~~three~~ valves is required because the combined

(Continued)

BASES

APPLICABILITY (continued) capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 when any all RCS cold leg temperatures are \leq the temperature below which LTOP is required as specified in the PTLR [275] $^{\circ}$ F or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head closure bolts fully de-tensioned.

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures \leq the temperature below which LTOP is required as specified in the PTLR [275] $^{\circ}$ F within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below the temperature below which LTOP is required as specified in the PTLR [275] $^{\circ}$ F, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. ~~Pressurizer safety valves are to be tested in accordance with the requirements of The ASME Code, Section XI of the ASME Code (Ref. 4), which provides the activities and frequencies necessary to satisfy the SRs, requires that safety and relief tests be performed in accordance with ANSI/ASME OM-a-1988 (Ref. 5). No additional requirements are specified. The surveillance specifies the lift settings to be within $\pm 1\%$ of nominal pressure of 2485 psig.~~

~~The pressurizer safety valve setpoint is $\pm [3]\%$ for OPERABILITY; however, the valves are reset to $\pm 1\%$ during the Surveillance to allow for drift.~~

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
 2. FSAR, Chapter-[15].
 3. WCAP-7769, Rev. 1, June 1972.
 4. ASME, Boiler and Pressure Vessel Code, Section XI.
 5. ~~Operation and Maintenance Code, 1987 with OM-a-1988 Addenda.~~
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases above their actuation setpoint and to close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

DCPP design includes three air operated pressurizer PORVs. Two of these PORVs have been designated as "Class I". These two valves provide the reactor vessel low temperature overpressure protection and they provide the means to depressurize the RCS following a steam generator tube rupture (SGTR). These functions must be accomplished under accident analyses assumptions such as loss of offsite power. Consequently, a Class I nitrogen backup system to the non-safety related air supply is provided for the two Class I PORVs. The identification of Class I is used to make a distinction between these two PORVs that must provide a safety-related function as opposed to the third remaining PORV that is designated as non-Class I. TS 3.4.12 for LTOP applies to the two Class I PORVs but not to the non-Class I PORV.

The non-Class I PORV is an element of the DCPP design for 100% load rejection without reactor trip. This valve is associated with plant transients as compared to accident mitigation. Although mitigation is not its primary purpose, the valve may be used for those functions also, although not credited for operation.

The three PORVs are the same design. The PORV that is not designated as Class I may be used, when instrument air is available, to control RCS pressure similarly to the Class I PORVs. However, two Class I PORVs satisfy the function with redundancy, therefore continued operation with the non-Class I PORV unavailable for RCS pressure control is allowed as long as the block valve or PORV can be closed to maintain the RCS pressure boundary. However, the plant capability to sustain a 100% load rejection without reactor trip would be compromised.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The three MOV block valves are the same design. The block valves are used to isolate the PORVs in case of excessive seat leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

(Continued)

BASES

BACKGROUND
(continued)

The PORVs may be manually cycled and are equipped with circuitry for automatic actuation. No credit is taken for PORV automatic actuation in the FSAR analyses for MODE 1, 2 or 3 transients where PORV operation may have a beneficial effect. Therefore the PORVs may be OPERABLE in either manual operation or the automatic mode. The automatic mode is the preferred configuration, as this provides pressure relieving capability without reliance on operator action.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permits performance of surveillances on the block valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated The PORV block valves are all powered from two separate safety trains (Ref. 1) vital busses.

The plant has ~~three~~ two PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure-High reactor trip setpoint up to and including the design step load decrease following a step reduction of 50% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and the two Class 1 PORVs are also used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

APPLICABLE
SAFETY
ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal or auxiliary pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. For the SGTR event, the PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

(Continued)

BASES (continued)

APPLICABLE
SAFETY
ANALYSES
(continued)

The PORVs are used in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical. By assuming PORV manual actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. Automatic actuation of the PORVs is not assumed in any design basis accidents during MODES 1, 2, and 3. Events that assume this condition include a turbine trip and the loss of normal feedwater (Ref. 2).

Pressurizer PORVs satisfy Criterion 3 of NRC Policy Statement 10 CFR 50.36 (c) (2) (ii)

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining the two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive seat leakage. Satisfying the LCO helps minimize challenges to fission product barriers. Note, however, that operability of the PORVs (as indicated by the surveillances) only requires that the PORVs be capable of being manually cycled to perform their safety function and that they need not be capable of automatic actuation since that is not a safety function.

APPLICABILITY

In MODES 1, 2, and 3, the PORVs are required to be OPERABLE to mitigate a SGTR and its block valve. The block valves are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs OPERABILITY in are also required to be OPERABLE in MODES 1, 2, and 3 to will also minimize challenges to the pressurizer safety valves.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. OPERABILITY of the PORVs requires them to be capable of manual operation. Automatic operation is not assumed in accident analyses and therefore is not a required safety function. The LCO 3.4.11 is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place and the reactor vessel head closure bolts not fully de-tensioned. LCO 3.4.12 addresses the PORV requirements in these MODES

(Continued)

BASES (continued)

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits ~~MODE changes with inoperable PORVs or block valves as one possible recourse to remaining in the Applicability of LCO 3.4.12. entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status. Testing is not performed in lower MODES.~~

A.1

~~With the PORVs may be inoperable and capable of being manually cycled, (e.g., excessive seat leakage) In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valves should be required to be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. No credit is given for automatic PORV operation in Reference 2 analyses for MODE 1, 2, and 3 transients. As such, the PORVs are considered OPERABLE in either manual control or in the automatic mode. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability isolation may be necessary due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use, and do not create a possibility for a small break LOCA. For these reasons, the block valve may be closed but the ACTION requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE and automatic actuation status prior to entering startup (MODE 2).~~

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

~~If one [or two] PORVs is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Time of~~

(Continued)

BASES (continued)

ACTIONS (continued) 1 hour ~~is~~ are reasonable, based on challenges to the PORVs during this time period, and provides the operator adequate time to correct the situation.

If the inoperable PORV valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one Class I PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status if it is Class I. If the valve is the non-Class I PORV, there is no required Completion Time. If the Class I PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply at least MODE 3 with Tavg less than 500°F, as required by Condition D.

C.1, C.2 and C.3

If one PORV block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The PORV control switch has three positions: open, close, and auto. Placing the PORV in manual control, if required in ACTION C, is accomplished by positioning the switch out of the auto control mode. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the associated PORV in manual control.

This action is taken to avoid the potential for a stuck open PORV if the valve were to open under automatic control at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. ~~preclude its automatic opening for an overpressure event and to. Because at least one Class I PORV remains OPERABLE, If the inoperable block valve is associated with a Class I PORV the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the Class I PORV block valve is based upon the Completion Time for restoring an inoperable Class I PORV in Condition B, since the PORVs are not capable of mitigating an overpressure event a SGTR when placed in manual control inoperable and not capable of being manually cycled. If the block valve is restored within the Completion Time of 72 hours, the power will be restored and the PORV restored to OPERABLE status will be transferred to the automatic mode of operation. If the block valve it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, at least MODE 3 with Tavg less than 500°F as required by Condition D.~~

If the inoperable block valve is associated with the non-Class I PORV, the block valve may be closed and the power removed. The 72 hour Completion Time for closing the block valve is the same applied in Required Action C.2. This recognizes that some restoration work may be required since the block valve is inoperable.

(Continued)

BASES (continued)

Restoration of the non-class I PORV block valve to OPERABLE status is not required because the non-Glass I PORV is not required to be available, although having the valve closed impairs the load rejection design capability. Therefore, once the block valve has been closed per Required Action C.3, Completion Time requirements of Condition D do not apply.

If the block valve can not be placed in the closed position, per Required Action C.3, Condition D applies and the unit must be taken to MODE 3 with Tavg less than 500°F until the block valve is restored or closed.

ACTIONS
(continued)

D.1, D.2 and D.3

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a MODE in which the LCO does not apply, condition below where the function of the PORVs to mitigate a SGTR event is needed. To achieve this status, the plant must be brought to at least MODE 3 with Tavg < 500°F within 6 hours and to MODE 4 within 12 hours. Additional action is required to be initiated immediately to continue efforts to restore the inoperable valve(s) to OPERABLE status. This action will ensure expedient measures are taken to reestablish OPERABLE PORVs and block valves while maintaining plant conditions above MODE 4 but less than 500°F. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4, and 5, and 6 with the reactor vessel head closure bolts not fully de-tensioned, maintaining Class I PORV OPERABILITY may be required by See LCO 3.4.12.

E.1, E.2, E.3, E.4 and E.5

If more than one Class I PORV is inoperable and not capable of being manually cycled, it is necessary to immediately initiate action to restore the valves either restore at least one valve and to, within the Completion Time of 1 hour, or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If one Class I PORV is restored and one Class I PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two [or three] Class I PORVs inoperable. If no Class I PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply, a condition below where the function of the PORVs to mitigate a SGTR event is not needed. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and reduce Tavg to < 500°F to MODE 4 within 12 hours.

Additional action is required to be initiated immediately to continue efforts to restore the inoperable valve(s) to OPERABLE status. This action will ensure expedient measures are taken to reestablish OPERABLE PORVs and block valves while maintaining plant conditions above MODE 4 but less than 500°F. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly

(Continued)

BASES (continued)

ACTIONS
(continued)

manner and without challenging plant systems. In MODES 4, and 5, and 6 with the reactor vessel head closure bolts not fully de-tensioned, maintaining Class I PORV OPERABILITY may be is required by See LCO 3.4.12.

F.1, F.2, F.3 and F.4

If more than one PORV block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control and restore at least one block valve within 2 hours and restore the remaining block valve within 72 hours. The PORV control switch has three positions, open, close and auto. Placing the PORV in manual control, if required in ACTION F, is accomplished by positioning the switch out of the auto control mode. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

If the inoperable block valve is associated with the non-Class I PORV, the block valve may be closed and the power removed. The 72 hour Completion Time for closing the block valve is the same used in Required Action F.3. This recognizes that some restoration work may be required since the block valve is inoperable. Restoration of the non-class I PORV block valve to OPERABLE status is not required because the non-Class I PORV is not required to be available, although having the valve closed impairs the load rejection design capability. Therefore, once the block valve has been closed per Required Action F.4, Completion Time requirements of Condition G do not apply.

If the block valve can not be placed in the closed position per Required Action F.4, Condition G applies and the unit must be taken to MODE 3 with T_{avg} less than 500°F until the block valve is restored or closed.

G.1, G.2 and G.3

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply condition below where the function of the PORVs to mitigate a SGTR event is not needed. To achieve this status, the plant must be brought to at least MODE 3 with $T_{avg} < 500°F$ within 6 hours and to MODE 4 within 12 hours. Additional action is required to be initiated immediately to continue efforts to restore the inoperable valve(s) to OPERABLE status. This action will ensure expedient measures are taken to reestablish OPERABLE PORVs and block valves while maintaining plant conditions above MODE 4 but less than 500°F. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4, and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.12, and 6 with the reactor vessel head closure bolts not fully de-tensioned, maintaining Class I PORV OPERABILITY is required by LCO 3.4.12.

SURVEILLANCE
REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the Frequency of 92 days is the ASME Code, Section XI O & M Code Part 10

(Continued)

BASES (continued)

(Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable Class I PORV that is incapable of being manually cycled, the maximum Completion Time to restore the Class I PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the Class I PORV to OPERABLE status (i.e., completion of the Required Actions fulfills the SR). If the block valve for a non-Class I PORV has been closed by Required Action C 3 or F 4, it can remain closed without further requirements for restoration. The non-Class I PORV is not required for accident mitigation and the closed block valve provides protection against a spurious opening of the PORV.

The Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed, in accordance with the Required Action of Condition A, B and E or Required Actions C 3 and F 4 this LCO.

Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in any MODE below MODE 2.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be manually actuated for mitigation of an SGTR. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the surveillance to be performed in any MODE below MODE 2.

SR 3.4.11.3

Verifying OPERABILITY of the safety related nitrogen supply for the Class I PORVs may be accomplished by:

- a. Isolating and venting the normal air supply, and
- b. Verifying that any leakage of the Class I backup nitrogen system is within its limits, and
- c. Operating the Class I PORVs through one complete cycle of full travel.

(Continued)

BASES (continued)

Operating the solenoid nitrogen control valves and check valves on the nitrogen supply system and operating the Class I PORVs through one complete cycle of full travel ensures the nitrogen backup supply for the Class I PORV operates properly when called upon. The Frequency of 18 months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate Class I PORV OPERABILITY.

SR 3.4.11.4 NOT USED

~~This Surveillance is not required for plants with permanent IE power supplies to the valves.~~

~~The Surveillance demonstrates that emergency power can be provided and is performed by transferring power from normal to emergency supply and cycling the valves. The Frequency of [18] months is based on a typical refueling cycle and industry accepted practice.~~

REFERENCES

1. Regulatory Guide 1.32, February 1977.
2. FSAR, Section 15.2.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Code for Operation and Maintenance of Nuclear Power Plants 1987 with 1988 Addenda, Part 10.
4. Generic Letter 90-06, "Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and generic issue 94, "Additional Low-Temperature Overpressure for Light-Water Reactors," Pursuant to 10CFR50.54(f), June 25, 1990

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The ~~PRESSURE TEMPERATURE LIMITS REPORT (PTLR)~~ provides the ~~maximum~~ allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperatures during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as ~~after~~ temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all ~~SI pumps but [one] [high pressure injection (HPI)] pump and one centrifugal charging pump (CCP)~~ incapable of injection into the RCS and isolating the accumulators.

~~Although not addressed in the LCO, the plant design also includes a positive displacement charging pump (PDP). Operation of a CCP or a PDP or simultaneous operation of both is controlled administratively. The pump operating combinations and limitations are discussed in B 3.4.12 ACTION B.1.~~

The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

(Continued)

BASES

**BACKGROUND
(continued)**

The pressurizer has three Power Operated Relief Valves (PORVs). Two of the three PORVs are classified as safety related and are designated for LTOP pressure protection. All the PORVs are air operated. These two safety related PORVs have a nitrogen gas backup to the non-safety related air supply.

The three PORVs are the same design. The PORV that is designated as non-Class I may be used, when Instrument air is available, to control RCS pressure similarly to the Class I PORVs although the non-Class I PORV does not receive an automatic open signal like the LTOP designated valves. Therefore, because no credit is taken for its operation for LTOP, continued operation with the non-Class I PORV unavailable for RCS pressure control is allowed as long as the associated block valve or non-Class I PORV can be closed to maintain the RCS pressure boundary.

In MODE 4 with the RHR loops in operation and in MODES 5 and 6, the operating RHR loop, connected to the RCS, can provide pressure relief capability through the RHR suction line relief valve. This capacity for RCS pressure relief is not assumed in the PTLR LTOP considerations and analyses and is not included in the LCO, ACTIONS, or Surveillances.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one [HPI or] charging pumps CCP for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

Additionally, CCPs in excess of the above limitations can be momentarily capable of injection into the RCS for swapping of inservice CCPs. This condition is acceptable based on the operator's attentiveness to RCS pressure during the pump switch over and the capability of the operator to limit a pressure increase.

The LTOP System for pressure relief consists of two Class I PORVs with reduced lift settings or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve, or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves Class I PORVs are required for redundancy. One RCS relief valve Class I PORV has adequate relieving capability to keep from prevent overpressurization for from the required allowable coolant input capability.

PORV Requirements

As designed for the LTOP System, each Class I PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic setpoint. The LTOP actuation logic monitors both RCS temperature and RCS

(Continued)

BASES

BACKGROUND
(continued)

pressure and determines when a condition not acceptable in the PTLR limits is approached. The evolution of RHR cooldown with no RCP forced circulation represents a condition where variation in RCS cold leg temperatures may occur. The RCS loop 2 and 3 wide range cold leg RCS temperature indications are auctioneered to select provide the lowest temperature input signal. Temperature indications from these two loops were selected to constitute a good representation of the overall four loop temperatures. However, in the event that only one RHR loop is in operation, temperature indications from RCS cold legs 2 and 3 will provide indication from a RCS loop into which the cooler water from the RHR discharge is entering. All four cold leg temperature indications are in the control room and provide a loop by loop comparison for the operator.

The LTOP system is placed into service and the block valves verified to be open by procedure at a RCS pressure of about 350 psig. This is an administrative action, not required by TS. However, if LTOP has not been placed into service prior to when the RCS temperature decreases to a temperature of about 323°F, 270°F, the LTOP enable alarm annunciates to alert the operator to place the LTOP system into service. Placing LTOP into service at this point is required to satisfy the LTOP Applicability requirements. Following being placed into service, LTOP will receive RCS temperature and pressure input. The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated PTLR LTOP pressure limit setpoint is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated LTOP value, and the temperature is lower than the enable temperature, a PORV is signaled to open. The two Class 1 PORVs operate individually with their own setpoints.

The PTLR specifies presents the PORV setpoints for LTOP. The setpoints are normally staggered so only one valve typically opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PORV is opened opens in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RHR Suction Relief Valve Requirements

During LTOP MODES, the RHR System is operated for decay heat removal and low pressure letdown control. Therefore, the RHR suction isolation valves are open in the piping from the RCS hot legs to the inlets of the RHR pumps. While these valves are open and the RHR suction valves are open, the RHR suction

(Continued)

BASES

BACKGROUND
(continued)

~~relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.~~

~~The RHR suction isolation valves and the RHR suction valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. Autoclosure interlocks are not permitted to cause the RHR suction isolation valves to close. The RHR suction relief valves are spring loaded, bellows type water relief valves with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.~~

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure ~~in an~~ during a RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

~~For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve, removing a PORV's internals, and disabling its block valve in the open position, or similarly establishing a vent by opening an RCS vent valve. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.~~

APPLICABLE
SAFETY
ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperatures exceeding ~~above the temperature below which LTOP is required as specified in the PTLR~~ [275]°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. ~~At or below the limiting temperature specified in the PTLR At about [275]°F and below,~~ overpressure prevention falls to two OPERABLE RCS relief valves Class I PORVs or to a depressurized RCS and a sufficiently sized RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition.

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSES
(continued)

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

- a. Inadvertent safety injection;
- b. Charging/letdown flow mismatch;
- c. ~~Accumulator discharge~~

Heat Input Type Transients)

- a. Inadvertent actuation of pressurizer heaters;
- b. Loss of RHR cooling; or
- c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

- a. Rendering all ~~SI but [one] [HPI] pumps and one CCP~~ incapable of injection;
- b. Deactivating the accumulator discharge isolation valves in their closed positions; and
- c. ~~Disallowing~~ ~~Precluding~~ start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop and ~~pressurizer water level is not less than 50%. LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection.~~

The Reference 4 analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only ~~one [HPI] pump and one charging pumps CCP~~ is actuated. Thus, the LCO allows only ~~one [HPI] pump [and one charging pumps] CCP~~ OPERABLE during the LTOP MODES. Since neither one RCS relief valve nor the RCS vent can handle the pressure transient ~~resulting~~ need from accumulator injection, when RCS temperature is low the LCO also requires ~~the~~ accumulators isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

(Continued)

BASES

APPLICABLE SAFETY ANALYSES (continued) The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. ~~The analyses show the effect of accumulator discharge is over a narrower RCS temperature range ([175]°F and below) than that of the LCO ([275]°F and below).~~

~~Fracture mechanics analyses established the~~ The current DCCP temperature of LTOP Applicability at of 270 [275]°F was determined in agreement with NRC Branch Technical Position 5-2. ~~This number was added to the current TS by LA 100/99~~

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6), requirements by having a maximum of ~~[one] [HPI] pump and one charging pumps]~~ ~~one CCP OPERABLE and SI actuation enabled.~~

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the P/T limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of ~~[one] [HPI] pump and one charging pumps]~~ ~~one CCP injecting into the RCS with the positive displacement charging pump (PDP) operating and with RCS letdown isolated.~~ These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met ~~at low temperature operation.~~

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. ~~The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," PTLR discusses these examinations.~~

~~The PORVs are considered active components. Thus, the~~ ~~The~~ failure of one ~~Class I~~ PORV is assumed to represent the worst case, single active failure.

RHR Suction Relief Valve Performance]

~~The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses must show that one RHR suction relief valve with a setpoint at or between [436.5] psig and [463.5] psig will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, an RHR suction relief valve will maintain RCS pressure to~~

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSES
(continued)

~~within the valve rated lift setpoint, plus an accumulation < 10% of the rated lift setpoint.~~

~~Although each RHR suction relief valve may itself meet single failure criteria, its inclusion and location within the RHR System does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for LTOP.~~

~~The RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.~~

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 2.07 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, ~~[one] HPI pump [and one charging pump] no SI pumps and one CCP~~ OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure. ~~The pathway from the RCS to the vent is also considered to be passive. The vent is considered to connect directly to the RCS. If the pathway includes devices with the potential to block the pathway, these devices must be secured to avoid blocking the vent.~~

The LTOP System satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c) (2) (ii).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when ~~the minimum RCS coolant input and pressure relief capabilities are OPERABLE within limits established in the LCO.~~ Violation of this LCO could lead to the loss of low temperature overpressure mitigation capability and violation of the Reference 1 PTLR limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires ~~that a maximum of [zero SI] [one] [HPI] pumps and [one CCP (except during pump swap operations)] charging pump be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized when when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.~~

(Continued)

BASES

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two RCS relief valves ~~Class 1 PORVs~~ as follows:

1. A ~~Class 1~~ PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

~~[2. Two OPERABLE RHR suction relief valves; or]~~

<p>An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valve and its RHR suction valve are open, its setpoint is at or between [436.5] psig and [463.5] psig, and testing has proven its ability to open at this setpoint.</p>	<p>An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valve and its RHR suction valve are open, its setpoint is at or between [436.5] psig and [463.5] psig, and testing has proven its ability to open at this setpoint.</p>
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<p>3- One OPERABLE PORV and one OPERABLE RHR suction relief valve; or</p>	<p>3- One OPERABLE PORV and one OPERABLE RHR suction relief valve; or</p>
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~~OR~~

b. ~~A depressurized RCS and an RCS vent.~~

An RCS vent is OPERABLE when open with an area of \geq ~~2.07~~ square inches.

~~Each~~ ~~Either~~ of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

~~the LCO is modified by a Note that permits two LCO capable of injecting into the RCS until 1 hour after completion of pump swap operation~~

APPLICABILITY This LCO is applicable in MODE 4 when any RCS cold leg temperature is \leq the temperature below which LTOP is required as specified in the PTLR [275]°F, in MODE 5, and in MODE 6 when the reactor vessel head is on and the vessel head closure bolts are not fully de-tensioned. RCS overpressure protection is not required in MODE 6 with the reactor vessel head closure bolts fully de-tensioned. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the limiting temperature specified in the PTLR [275]°F. When the reactor vessel head is off, overpressurization cannot occur.

~~LCO 3.4.3~~ The PTLR provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, 3, and MODE 4 above the limiting RCS temperature specified in the PTLR [275]°F.

(Continued)

BASES (continued)

APPLICABILITY (continued) Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows ~~is available for~~ operator action to mitigate the event.

The Applicability is modified by ~~two Notes~~. ~~Note 1 states~~ stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions. ~~Note 2 states that more than one charging pump may be capable of injection into the RCS during, and up to 1 hour after swapping charging pump operation.~~

ACTIONS

A.1 and B.1

With ~~one two~~ or more ~~SI HPI pumps or two CCPs~~ capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

~~Required Action B.1 is modified by a Note that permits two charging pumps capable of RCS injection for < 15 minutes to allow for pump swaps.~~

~~The CCP and the PDP are capable of injecting into the RCS both operating alone or simultaneously. Their operation is limited to the conditions specified in the PTLR. The current limitations are based on RCS temperature as follows:~~

<u>RCS Temperature</u>	<u>Allowable Charging Pumps Capable of Injecting into the RCS</u>
<u>Greater than 270°F</u>	<u>Two CCPs AND one PDP</u>
<u>Less than or equal to 270°F but greater than approximately 162°F</u>	<u>One CCP AND one PDP</u>
<u>Less than or equal to approximately 162°F but greater than approximately 134°F</u>	<u>One CCP OR one PDP</u>
<u>Less than or equal to approximately 134°F</u>	<u>One CCP OR one PDP AND ECCS charging injection flow path isolated</u>

(Continued)

BASES

If the RCS reactor vessel head is fully de-tensioned or the RCS is not intact, the above limitation on charging and SI pumps do not apply.

ACTIONS
(continued)

C.1, D.1, and D.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required ACTION D.1 and Required ACTION D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to $>$ the temperature below which LTOP is required as specified in the PTLR [275]^oF, an accumulator pressure of 600 psig cannot exceed the LTOP P/T limits if the accumulators are fully injected. The second option to depressurize the accumulators below the LTOP P/T limits from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

E.1

In MODE 4 when any RCS cold leg temperature is \leq the temperature below which LTOP is required as specified in the PTLR [275]^oF, with one required RCS Class I PORV relief valve inoperable, the RCS relief valve Class I PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS Class I PORVs relief valves [in any combination of the PORVs and the RHR suction relief valves] are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the RCS relief valves Class I PORVs is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

F.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves Class I PORVs inoperable in MODE 5 or in MODE 6 with the head on and the vessel head closure bolts not fully de-tensioned, the Completion Time to restore two valves to OPERABLE status is 24 hours.

(Continued)

BASES

ACTIONS
 (continued)

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve Class I PORV to protect against overpressure events.

G.1

The RCS must be depressurized and a vent must be established within 8 hours when:

- a. Both required RCS relief valves Class I PORVs are inoperable; or
- b. A Required Action and associated Completion Time of Condition A, B, D, E, or F is not met; or
- c. The LTOP System is inoperable for any reason other than Condition A, B, C, D, E, or F.

The vent must be sized ≥ 2.07 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE
 REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2 and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of zero SI [one] [HPI] pumps [and a maximum of one charging pumps] CCP are verified incapable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked-out, their breakers open. Verification that each accumulator is isolated is only required when accumulator isolation is required as stated in Note 1 to the Applicability.

The SI [HPI] pumps and one CCP charging pump[s] are rendered incapable of injecting into the RCS for example, through removing the power from the pumps by racking the breakers out under administrative control or by isolating the discharge of the pump by closed isolation valves with power removed from the operators or by a manual isolation valve secured in the closed position.

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

An alternate method of LTOP providing low temperature overpressure protection control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not that could result in an injection into the RCS. This may be accomplished through the pump control switch being placed in [pull to lock] and at least one valve in the discharge flow path being closed. An inoperable pump may be energized for test or for accumulator fill provided the discharge of the pump is isolated from the RCS by closed isolation valve(s) with power removed from the valve operator(s), or by manual isolation valve(s) sealed in the closed position. The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.4 NOT USED

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.7 for the RHR suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.~~

~~The RHR suction valve is verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valve remains open.~~

~~The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.~~

SR 3.4.12.5

The RCS vent of ≥ 207 square inches is proven OPERABLE by verifying its open condition either:

- a. Once every 12 hours for a valve that cannot be is not locked, sealed, or secured in the open position.
- b. Once every 31 days for other vent paths (e.g., a valve that is locked, sealed, or otherwise secured in the open position. A removed pressurizer safety valve or open manway also fits this category.

The Any passive vent path arrangement need must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12.b.

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.12.6

The ~~Class I~~ PORV block valve must be verified open every 72 hours to provide the flow path for each required ~~Class I~~ PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. ~~This surveillance is performed if the PORV satisfies the LCO.~~

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the ~~Class I~~ PORV block valve remains open.

SR 3.4.12.7 ~~NOT USED~~

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.4 for the RHR suction valve Surveillance and for a description of the requirements of the Inservice Testing Program.) This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.~~

~~Every 31 days the RHR suction isolation valve is verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The "locked open" valve must be locally verified in its open position with the manual actuator locked in its inactive position. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve position.~~

SR 3.4.12.8

~~The Note SR states that the SR is not required to be performed until 12 hours after decreasing any RCS cold leg temperature to $\leq 275^{\circ}\text{F}$ the temperature below which LTOP is required as specified in the PTLR. The test must be performed within 12 hours after entering the LTOP MODES. The 12 hour allowance Frequency considers the unlikelihood of a low temperature overpressure event during this time.~~

~~Following the initial 12 hour SR, while remaining in the Applicable LTOP MODE, the SR will be performed every 31 days thereafter on each required Class I PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed limits in the PTLR. PORV actuation could depressurize the RCS and is not required.~~

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

~~Performance of a COT is required within 12 hours after decreasing RCS temperature to < the temperature below which LTOP is required as specified in the PTLR [275]°F and every 31 days thereafter on each required Class I PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.~~

~~A Note has been added indicating that this SR is required to be performed met 12 hours after decreasing RCS cold leg temperature to < the temperature below which LTOP is required as specified by the PTLR [275]°F. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.~~

SR 3.4.12.9

Performance of a CHANNEL CALIBRATION on each required Class I PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ~~ASME, Boiler and Pressure Vessel Code, Section III. NOT USED~~
4. FSAR, Chapter 5.
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.
8. ~~ASME, Boiler and Pressure Vessel Code, Section XI. NOT USED~~

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

Possible leakage from a Control Rod Drive Mechanism (CRDM) canopy seal weld may be construed as either identified or unidentified LEAKAGE but not construed as pressure boundary LEAKAGE in accordance with Westinghouse letter PGE-88-622.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leak tight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to

(Continued)

BASES

APPLICABLE SAFETY ANALYSES (continued) the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

~~The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.~~

~~The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 (Ref. 6) or the staff approved licensing basis (i.e., a small fraction of these limits).~~

~~The SGTR (Ref. 3) is more limiting for radiological releases at the site boundary. The radiological dose analysis assumes loss of off-site power at the time of reactor trip with no subsequent condenser cooling available. The steam generator (SG) PORV for the SG that has sustained the tube rupture is assumed to be open for 30 minutes, at which time the RCS pressure is below the lift setting of the PORV. The dose consequences resulting from the SGTR accident are within the limits defined in 10 CFR 100 (Ref. 6).~~

~~The safety analysis for RCS main loop piping for GDC-4 (Ref. 1) assumes 1 gpm unidentified leakage and monitoring per RG 1.45 (Ref. 2) are maintained (Ref. 4 and 5).~~

The RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement. ~~10 CFR 50.36 (c) (2) (11).~~

LCO RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets, or the CRDM canopy seal welds is not pressure boundary LEAKAGE.

Pressure boundary leakage is defined as "non-isolable" leakage. A "non-isolable" RCS leak is one that is not capable of being isolated from the RCS using installed automatic or accessible manual valves.

(Continued)

BASES

LCO
(continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified identified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Identified LEAKAGE does not include LEAKAGE from portions of the Chemical and Volume Control System outside of containment that can be isolated from the RCS. LEAKAGE of this nature may be reviewed for possible impact on the Primary Coolant Sources Outside Containment Program. Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through Any One SG

The 500 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative

APPLICABILITY In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series

(Continued)

BASES

APPLICABILITY (continued) in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. Therefore, a Note is added allowing that this SR is not required to be performed in MODES 3 and 4 until 12 hours of after establishing steady state operation near operating pressure, have been established. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. and a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, (avg changes less than 5°F per hour) power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and by the containment structure sump level and flow monitoring system. It should be noted that LEAKAGE past seals, and gaskets or CRDM canopy seal welds is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be performed during steady state operation. The 12 hour Frequency after steady state operation has been achieved provides for those situations following a transient such that the 72 hours plus extension allowed by SR 3.0.2 would be exceeded. Under these circumstances, the SR would be due within 12 hours after steady state operation has been reestablished and every 72 hours thereafter during steady state operation. Steady state is defined as T_{avg} being changed by less than 5°F/hour.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions. This surveillance does not tie directly to any of the leakage criteria in the LCO or of the conditions, therefore failure to meet this surveillance is considered failure to meet the integrity goals of the LCO and LCO 3.0.3 applies.

(Continued)

BASES

- REFERENCES
1. 10 CFR 50, Appendix A, GDC 4 and 30-30.
 2. Regulatory Guide 1.45, May 1973.
 3. FSAR, Section 15.
 4. FSAR, Section 3
 5. NUREG-1601, Volume 3, November, 1984
 6. 10 CFR 100
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND 10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leak tight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The Exceeding the leakage limit is an indication that may indicate the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

- a. Residual Heat Removal (RHR) System;
- b. Safety Injection System; and
- c. Chemical and Volume Control System.

(Continued)

BASES

BACKGROUND (continued) Violation of this LCO could result in continued degradation of a PIV, which could leak to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement. ~~10 CFR 50.36 (c) (2) (11)~~.

LCO RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. ~~The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.~~

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

(Continued)

BASES

APPLICABILITY In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment

ACTIONS

The ACTIONS are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1, A.2.1 and A.2.2

The flow path must be isolated by two valves. Required ACTIONS A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB or the high pressure portion of the system

Required ACTION A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required ACTION A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring the RCS PIV to within limits. The 72 hours Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This time frame considers the time required to complete the ACTION and the low probability of a second valve failing during this time period

or

~~The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This time frame considers the time required to complete this Action and the low probability of a second valve failing during this period. (Reviewer Note: Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.)~~

(Continued)

BASES

ACTIONS
(continued) B.1 and B.2

If leakage cannot be reduced, ~~the system isolated~~ or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant system.

C-1

~~The inoperability of the RHR autoclosure interlock renders the RHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the RHR autoclosure interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the autoclosure.~~

SURVEILLANCE
REQUIREMENTS SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. ~~This method results in testing each valve separately.~~ If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to ~~be~~ performed every ~~18~~ months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The ~~18~~ month frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) QSM Code, Section XI Part 10 (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(Continued)

BASES

SURVEILLANCE REQUIREMENTS SR 3.4.14.1 (continued)

~~Test pressures less than 2235 psig but greater than 150 psig are allowed for valves where higher pressures would tend to diminish leakage channel opening. Observed leakage shall be adjusted for actual pressure to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one half power.~~

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been resealed. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. ~~The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months.~~ In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

~~Testing is not required for the RHR suction isolation valves more frequently than 18 months as these valves are motor operated with control room position indication, inadvertent opening interlocks and system high pressure alarms.~~

SR 3.4.14.2 and 3.4.14.3 NOT USED

~~Verifying that the RHR system autoclosure interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR heat exchangers beyond 125% of their design pressure of [600] psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be <[425] psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The [18] month frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.~~

(Continued)

BASES

~~These SRS are modified by Notes allowing the RHR autoclosure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.~~

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- REFERENCES
1. 10 CFR 50.2.
 2. 10 CFR 50.55a(c).
 3. 10 CFR 50, Appendix A, Section V, GDC 55.
 4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
 5. NUREG-0677, May 1980.
 6. ~~NOT USED [Document containing list of PIVs.]~~
 7. ~~ASME, Boiler and Pressure Vessel Code, Section XI, Code for Operation and Maintenance of Nuclear Power Plants 1987, with 1988 Addenda, Part 10.~~
 8. 10 CFR 50.55a(g).
-
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sumps used to collect unidentified LEAKAGE ~~is (or) and the containment fan cooling unit (CFCU) condensate collection monitor are instrumented to alarm for~~ capable of detecting increases of 0.5 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

~~Each CFCU has an individual condensate collection monitor. The condensate from the cooling coils passes out from the CFCU to a containment sump. The condensate collection system design does not use an on-line flow monitor. The condensate drain flow can be collected, measured, and then using the elapsed time of the collection, the average flow rate can be determined. This monitoring can be done from the control room. Although multiple CFCUs may be operating, any individual CFCU condensate monitor may be employed to provide indication of the condensate flow rate.~~

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} $\mu\text{Ci/cc}$ radioactivity for particulate monitoring and of 10^{-6} $\mu\text{Ci/cc}$ radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

(Continued)

BASES

BACKGROUND
(continued)

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew point is well within the sensitivity range of available instruments.

Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump [and condensate flow from air coolers]. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO

APPLICABLE
SAFETY
ANALYSES

The asymmetric loads produced by postulated breaks are the result of assumed pressure imbalance, both internal and external to the RCS. The internal asymmetric loads result from a rapid decompression that causes large transient pressure differentials across the core barrel and fuel assemblies. The external asymmetric loads result from the rapid depressurization of the annulus regions, such as the annulus between the reactor vessel and the shield wall, and cause large transient pressure differentials to act on the vessel. These differential pressure loads could damage RCS supports, core cooling equipment or core internals. This concern was first identified as Multiplant Action (MPA) D-10 and subsequently as Unresolved Safety Issue (USI) 2, "Asymmetric LOCA Loads" (Ref. 4).

The resolution of USI-2 for Westinghouse PWRs was the use of fracture mechanics technology for RCS piping > 10 inches diameter. (Ref. 5). This technology became known as leak before break (LBB). Included within the LBB methodology was the requirement to have leak detection systems capable of detecting a 10 gpm leak within four hours. This leakage rate is designed to ensure that adequate margins exist to detect leaks in a timely manner during normal operating conditions. The use of the LBB methodology is described in Reference 6.

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described

(Continued)

BASES

APPLICABLE SAFETY in the FSAR (Ref. 3). ~~Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its~~
ANALYSES (continued) ~~source to an instrument location yields an acceptable overall response time.~~

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary.

Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur ~~that could be~~ detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of ~~the NRC Policy Statement.~~ ~~10 CFR 50.36 (c) (2) (11)~~

LCO One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitoring systems, ~~in combination with a gaseous or the particulate radioactivity monitor and either a CFCU condensate collection monitor or a gaseous radioactivity monitor~~ provides an acceptable minimum

APPLICABILITY Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE. In MODE 5 or 6, the temperature is to be $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and

ACTIONS ~~ACTIONS are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required containment sump monitor, the required atmospheric particulate monitor, the required atmospheric gaseous monitor or the required CFCU condensate collection monitor are inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.~~

(Continued)

BASES

ACTIONS (continued) A.1 and A.2

With the required containment sump monitors inoperable, ~~no other form of sampling can provide the equivalent information; however,~~ RCS water inventory balance, the containment atmosphere particulate radioactivity monitor, and the CFCU condensate collection monitoring system will provide indications of changes in leakage. Together with the atmosphere monitors, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (near rated operating pressure with stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required sump monitoring system to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitoring system failure. This time is acceptable considering the Frequency and adequacy of the RCS water inventory balance required by Required ACTION A.1.

~~Required ACTION A.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required containment sump monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.~~

B.1.1, B.1.2, and B.2.1, and B.2.2

With ~~the~~ both gaseous and particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere particulate radioactivity monitors. ~~Alternatively, continued operation is allowed if the air cooling condensate flow rate monitoring system is OPERABLE, provided grab samples are taken or water inventory balances are performed every 24 hours.~~

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (with stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of LEAKAGE detection is available.

(Continued)

BASES

ACTIONS
(continued)

~~Required Action B.1 and Required Action B.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required gaseous and particulate containment atmosphere radioactivity monitor channel is inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.~~

C.1 and C.2

~~With the required containment air cooling condensate flow rate monitor inoperable, alternative action is again required. Either SR 3.4.15.1 must be performed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. Provided a CHANNEL CHECK is performed every 8 hours or a water inventory balance is performed every 24 hours, reactor operation may continue while awaiting restoration of the containment air cooling condensate flow rate monitor to OPERABLE status.~~

~~The 24 hour interval provides periodic information that is adequate to detect RCS LEAKAGE.~~

C.1.1, C.1.2, C.2.1, and C.2.2

~~With the required containment atmosphere gaseous radioactivity monitor and the required CFCU condensate collection monitor inoperable, the means of detecting leakage are the containment sump monitoring system and the containment atmosphere particulate radioactivity monitor. This Condition does not provide all the required diverse means of leakage detection. With both gaseous containment atmosphere radioactivity monitoring and CFCU condensate monitoring instrumentation channels inoperable, alternate action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.~~

The ~~Follow-up~~ Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

(Continued)

BASES

DE.1 and DE.2

ACTIONS
(continued)

If a Required Action of Condition A, B, or C, or D cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

EF.1

With all required monitors inoperable, (LCO a, b, and c) no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required. With two of the three groups of leak detection monitoring not operable, the two groups will enter their respective ACTION and Completion statements. The third group provides a continued monitoring function.

SURVEILLANCE REQUIREMENTS SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence that the channels ~~is~~ are operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off-normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a ~~CHANNEL FUNCTIONAL TEST (CFT) CFT~~ on the required containment atmosphere radioactivity monitors. The test ensures that the monitors can perform ~~its~~ their function in the desired manner including alarm functions. ~~The test verifies the alarm setpoint and relative accuracy of the instrument string setting.~~ The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3, SR 3.4.15.4, and SR 3.4.15.5

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

(Continued)

BASES

- REFERENCES
1. 10 CFR 50, Appendix A, Section IV, GDC 30.
 2. Regulatory Guide 1.45.
 3. FSAR, Section 5.2.7.
 4. NUREG-609, "Asymmetric Blowdown Loads on PWR Primary System" 1981
 5. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Breaks in PWR Pipe Breaks in PWR Primary Main Loops"
 6. FSAR, Section 3.6B.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation

**APPLICABLE
SAFETY
ANALYSES**

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 from LCO 3.7.18.6, "Secondary Specific Activity."

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1%

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSES
(continued)

failed fuel, which closely equals the LCO limit of $100/E$ $\mu\text{Ci/gm}$ for gross specific activity.

The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to $60.0 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement. ~~10 CFR 50.36 (c) (2) (11)~~

LCO

The specific iodine activity is limited to $1.0 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of $\mu\text{Ci/gm}$ equal to 100 divided by E (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

(Continued)

BASES

APPLICABILITY In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq 500^{\circ}\text{F}$, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature $< 500^{\circ}\text{F}$, and in MODES 4 and 5, the offsite release of radioactivity from the affected SG in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety and relief valves

ACTIONS

A.1 and A.2

A Note to these ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required ~~allowed~~ to obtain and analyze a sample. Sampling is ~~done to~~ continued to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is ~~required, allowed to~~ permit recovery, if the limit violation resulted from normal iodine spiking.

B.1 and B.2

With the gross specific activity in excess of the allowed limit, ~~an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample. The unit must be placed in a MODE in which the requirement does not apply.~~

The change within 6 hours to MODE 3 and RCS average temperature $< 500^{\circ}\text{F}$ lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the affected SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

(Continued)

BASES

C.1 and C.2

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems

SURVEILLANCE
REQUIREMENTSSR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken, beta-gamma activity and the total of all identified gaseous activities in the sample within two hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change $\geq 15\%$ RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate less indicative results.

SR 3.4.16.3

A radiochemical analysis for \bar{E} determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium (as defined in SR 3.4.16.3 NOTE) conditions. The \bar{E} determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for \bar{E} is a measurement of the average energies per disintegration

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

~~for isotopes with half lives longer than 15 minutes, excluding iodines. The qualitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 15 minutes and all radiiodines which are identified in the reactor coolant. The specific activity for these individual radionuclides shall be used in the determination of E for the reactor coolant sample. Determination of the contributors to E shall be based upon those energy peaks identifiable with a 95% confidence level. The Frequency of 184 days recognizes E does not change rapidly.~~

This SR has been modified by a Note that indicates sampling for E determination is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for E is representative and not skewed by a crud burst or other similar abnormal event

REFERENCES

1. 10 CFR 100.11, 1973.
 2. FSAR, Sections 15.4.3 and 15.5.20
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~~B-3.4 REACTOR COOLANT SYSTEM (RCS)~~

~~B-3.4.17 RCS Loop Isolation Valves~~

~~BASES~~

~~BACKGROUND The RCS may be operated with loops isolated in order to perform maintenance. While operating with a loop isolated, there is potential for inadvertently opening the isolation valves in the isolated loop. In this event, the coolant in the isolated loop would suddenly begin to mix with the coolant in the operating loops. This situation has the potential of causing a positive reactivity addition with a corresponding reduction of SDM if:~~

- ~~a. The temperature in the isolated loop is lower than the temperature in the operating loops (cold water incident); or~~
- ~~b. The boron concentration in the isolated loop is lower than the boron concentration in the operating loops (boron dilution incident).~~

~~As discussed in the FSAR (Ref. 1), the startup of an isolated loop is performed in a controlled manner that virtually eliminates any sudden positive reactivity addition from cold water and/or boron dilution because:~~

- ~~a. LCO 3.4.18, "RCS Isolated Loop Startup," and plant operating procedures require that the boron concentration in the isolated loop be maintained higher than the boron concentration of the operating loops, thus eliminating the potential for introducing coolant from the isolated loop that could dilute the boron concentration in the operating loops;~~
- ~~b. The cold leg loop isolation valve cannot be opened unless the temperatures of both the hot and cold legs of the isolated loop are within 20°F of the temperatures of the hot and cold legs of the operating loops (compliance is ensured by operating procedures and automatic interlocks); and~~
- ~~c. Other automatic interlocks, all of which are part of the Reactor Protection System (RPS), prevent opening the hot leg loop isolation valve unless the cold leg loop isolation valve is fully closed~~

~~APPLICABLE SAFETY ANALYSES During startup of an isolated loop in accordance with LCO 3.4.18, the cold leg loop isolation valve interlocks and operating procedures prevent opening of the valve until the isolated loop and operating loop boron concentrations and temperatures are equalized. This ensures that any undesirable reactivity effect from the isolated loop does not occur.~~

~~The safety analyses assume a minimum SDM as an initial condition for Design Basis Accidents (DBAs) (Ref. 1). Violation of the LCO, combined with mixing of the isolated loop coolant into the operating loops, could result in the SDM being less than that assumed in the safety analyses.~~

(Continued)

BASES

~~The above analyses are for DBAs that establish the acceptance limits for the RCS loop isolation valves. Reference to the analyses for these DBAs is used to assess changes to the RCS loop isolation valves as they relate to the acceptance limits.~~

~~The boron concentration of an isolated loop may affect SDM and therefore RCS loop isolation valves satisfy Criterion 2 of the NRC Policy Statement.~~

LCO

~~This LCO ensures that a loop isolation valve that becomes closed in MODES 1 through 4 is fully isolated and the plant placed in MODE 5. Loop isolation valves are used for performing maintenance when the plant is in MODE 5 or 6, and startup of an isolated loop is covered by LCO 3.4.18.~~

APPLICABILITY

~~In MODES 1 through 4, this LCO is applicable when unisolating an isolated loop with a boron concentration less than that of the operating loops may cause an inadvertent criticality.~~

~~In MODES 5 and 6, the SDM of the operating loops is large enough to permit operation with isolated loops. In these MODES, controlled startup of isolated loops is possible without significant risk of inadvertent criticality.~~

ACTIONS

~~The Actions have been provided with a Note to clarify that all RCS loop isolation valves for this LCO are treated as separate entities, each with separate Completion Times, i.e., the Completion Time is on a component basis.~~

A-1

~~If power is inadvertently restored to one or more loop isolation valve operators, the potential exists for accidental isolation of a loop with a subsequent inadvertent startup of the isolated loop. The loop isolation valves have motor operators. Therefore, these valves will maintain their last position when power is removed from the valve operator. With power applied to the valve operators, only the interlocks prevent the valve from being operated. Although operating procedures and interlocks make the occurrence of this event unlikely, the prudent action is to remove power from the loop isolation valve operators. The Completion Time of 30 minutes to remove power from the loop isolation valve operators is sufficient considering the complexity of the task.~~

(Continued)

BASES

B.1, B.2, and B.3

~~Should a loop isolation valve be closed in MODES 1 through 4, the affected loop must be fully isolated immediately and the plant placed in MODE 5 to preclude inadvertent startup of the loop and the potential inadvertent criticality. The Completion Time of Required Action B.1 allows time for borating the operating loops to a shutdown boration level such that the plant can be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

SURVEILLANCE
REQUIREMENTS

SR 3.4.17.1

~~The Surveillance is performed at least once per 31 days to ensure that the RCS loop isolation valves are open, with power removed from the loop isolation valve operators. The primary function of this Surveillance is to ensure that power is removed from the valve operators, since SR 3.4.4.1 of LCO 3.4.4, "RCS Loops MODES 1 and 2," ensures that the loop isolation valves are open by verifying every 12 hours that all loops are operating and circulating reactor coolant. The Frequency of 31 days ensures that the required flow can be made available, is based on engineering judgment, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 31 day Frequency is justified.~~

REFERENCES

1. FSAR, Section [15.2.6].

~~B 3.4 REACTOR COOLANT SYSTEM (RCS)~~

~~B 3.4.18 RCS Isolated Loop Startup~~

BASES

~~BACKGROUND The RCS may be operated with loops isolated in MODES 5 and 6 in order to perform maintenance. While operating with a loop isolated, there is potential for inadvertently opening the isolation valves in the isolated loop. In this event, the coolant in the isolated loop would suddenly begin to mix with the coolant in the operating loops. This situation has the potential of causing a positive reactivity addition with a corresponding reduction of SDM if~~

- ~~a. The temperature in the isolated loop is lower than the temperature in the operating loops (cold water incident); or~~
- ~~b. The boron concentration in the isolated loop is lower than the boron concentration in the operating loops (boron dilution incident).~~

~~As discussed in the FSAR (Ref. 1), the startup of an isolated loop is done in a controlled manner that virtually eliminates any sudden reactivity addition from cold water or boron dilution because:~~

- ~~a. This LCO and plant operating procedures require that the boron concentration in the isolated loop be maintained higher than the boron concentration of the operating loops, thus eliminating the potential for introducing coolant from the isolated loop that could dilute the boron concentration in the operating loops.~~
- ~~b. The cold leg loop isolation valve cannot be opened unless the temperatures of both the hot leg and cold leg of the isolated loop are within 20°F of the operating loops. Compliance with the temperature requirement is ensured by operating procedures and automatic interlocks.~~
- ~~c. Other automatic interlocks prevent opening the hot leg loop isolation valve unless the cold leg loop isolation valve is fully closed. All of the interlocks are part of the Reactor Protection System~~

~~APPLICABLE SAFETY ANALYSES During startup of an isolated loop, the cold leg loop isolation valve interlocks and operating procedures prevent opening the valve until the isolated loop and operating loop boron concentrations and temperatures are equalized. This ensures that any undesirable reactivity effect from the isolated loop does not occur.~~

~~The safety analyses assume a minimum SDM as an initial condition for Design Basis Accidents. Violation of this LCO could result in the SDM being reduced in the operating loops to less than that assumed in the safety analyses.~~

(Continued)

BASES

~~The boron concentration of an isolated loop may affect SDM and therefore RCS isolated loop startup satisfies Criterion 2 of the NRC Policy Statement~~

LCO ~~Loop isolation valves are used for performing maintenance when the plant is in MODE 5 or 6. This LCO ensures that the loop isolation valves remain closed until the differentials of temperature and boron concentration between the operating loops and the isolated loops are within acceptable limits.~~

APPLICABILITY ~~In MODES 5 and 6, the SDM of the operating loops is large enough to permit operation with isolated loops. Controlled startup of isolated loops is possible without significant risk of inadvertent criticality. This LCO is applicable under these conditions.~~

ACTIONS ~~A.1 and A.2~~

~~Required Action A.1 and Required Action A.2 assume that the prerequisites of the LCO are not met and a loop isolation valve has been inadvertently opened. Therefore, the Actions require immediate closure of isolation valves to preclude a boron dilution event or a cold water event. However, each Required Action is preceded by a Note that states that Action is required only when a specific concentration or temperature requirement is not met~~

SURVEILLANCE REQUIREMENTS ~~SR 3.4.18.1~~

~~This Surveillance is performed to ensure that the temperature differential between the isolated loop and the operating loops is $\leq [20]^{\circ}\text{F}$. Performing the Surveillance 30 minutes prior to opening the cold leg isolation valve in the isolated loop provides reasonable assurance, based on engineering judgment, that the temperature differential will stay within limits until the cold leg isolation valve is opened. This Frequency has been shown to be acceptable through operating experience.~~

~~SR 3.4.18.2~~

~~To ensure that the boron concentration of the isolated loop is greater than or equal to the boron concentration of the operating loops, a Surveillance is performed 2 hours prior to opening either the hot or cold leg isolation valve. Performing the Surveillance 2 hours prior to opening either the hot or cold leg isolation valve provides reasonable assurance the boron concentration difference will stay within acceptable limits until the loop is unisolated. This Frequency has been shown to be acceptable through operating experience.~~

REFERENCES ~~1. FSAR, Section [15.2.6].~~

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.

Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.

Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.

Methodology For Mark-up of NUREG-1431 Specifications

Bracket Inserts -

The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(7 Pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.4

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER

JUSTIFICATION

- 3.4-01 This change clarifies intent of wording for the allowance to remove pumps from operation by changing "de-energized" to "removed from operation" consistent with Industry Traveler TSTF-153.
- 3.4-02 This change revises Condition A of ITS 3.4.6 to cover any required loop's inoperability and adds Required Action A.2 indicating that cooldown to MODE 5 is only required if an RHR loop is OPERABLE. Consistent with the CTS, this change addresses all possible configurations under the LCO and clarifies potentially confusing ISTS language with regard to the number of loops that are required. Condition B is deleted.
- 3.4-03 The CTS allows 1 hour for all RHR pumps to be removed from operation. One hour is the time allowed in the version of the Westinghouse STS from which the plant-specific TS was prepared. The change and its basis are not discussed in the documentation for the improved STS. Plant-specific experience has not indicated that the one hour allowance is unsafe or unacceptable. This is the current licensing basis.
- 3.4-04 The symbol ">" in LCO 3.4.8, NOTES 1.a. is replaced with the words "at least". This is consistent with the use of "at least" in CTS 3.4.1.2 footnote *, 3.4.1.3 footnote *, 3.4.1.4.1 footnote *), and in ITS 3.4.6 LCO, and 3.4.7 LCO. The change makes the ITS consistent with the CTS.
- 3.4-05 Not applicable to Diablo Canyon Power Plant (DCPP.) See Conversion Comparison Table (Enclosure 6B).
- 3.4-06 These changes are consistent with the current plant specific analysis and are reflective of the CTS. The plant-specific analysis does not allow injection from any safety injection pumps and does allow injection from [one] centrifugal charging pumps. The changes are made consistently in the LCO, ACTIONS, and Surveillance Requirements.
- 3.4-07 The word "all" in the improved TS 3.4.12 Applicability is replaced by the word "any". Use of "any" makes the Applicability consistent with the Bases. The intention is to enter LTOP Applicability if one or more of the cold legs have a temperature below the limit specified in the PTLR. Similar statements appear in ITS 3.5.2, 3.4.6, 3.4.7, and 3.4.10, all of which are consistent in that all cold legs must be above the PTLR specified temperature to conduct various evolutions whereas any cold leg below that temperature initiates the need for vessel low temperature overpressure protection. The change makes the ITS consistent with the CTS.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.4

<u>CHANGE NUMBER</u>	<u>JUSTIFICATION</u>
3.4-08	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.4-09	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.4-10	The minimum temperature specified in degrees below which the RCS must not be subject to low temperature overpressurization is replaced by the statement "the temperature below which LTOP is required as specified in the PTLR," per Industry Traveler WOG-67, Rev 1. The pressure temperature limits report (PTLR) provides the limiting temperature that is a plant-specific requirement based on the reactor vessel material characteristics documented in the PTLR and is periodically reviewed and revised. Referring to the PTLR for the current value is consistent with the relocation of the pressure temperature limits from improved TS Section 3.4.3 to the PTLR. This is an administrative change.
3.4-11	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.4-12	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.4-13	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.4-14	This change revises grouping of required detectors to be consistent with the CTS and with Reg. Guide 1.45, Regulatory Position C.3.
3.4-15	LCO 3.0.4 is applicable to Condition [D] of ITS 3.4.15 (required containment atmosphere gaseous radioactivity monitor and containment cooler condensate monitoring system inoperable) because other mechanisms (i.e., grab samples, RCS inventory balance, containment sump monitor, etc.) exist which are capable of adequately detecting RCS leakage and because a 30 day AOT is usually accompanied by a LCO 3.0.4 exception (e.g., PAM and Remote Shutdown Technical Specifications). Therefore, an exception to LCO 3.0.4 was added to Condition [D]. As there is already an LCO 3.0.4 exception to Conditions A and B [and LCO 3.0.4 is not applicable to ACTION C (which allows indefinite operation),] the LCO 3.0.4 exception was moved to apply to all ACTIONS and the specific exceptions for Conditions A and B were deleted. The change is made per Industry Traveler TSTF-60.
3.4-16	Consistent with CTS requirements [and with Condition B,] a requirement to perform 24 hour containment atmosphere samples is added when a gaseous monitor is inoperable []. Also, consistent with Conditions A and B, the performance of SR 3.4.13.1 every 24 hours as an alternative to analyzing a containment atmosphere sample is allowed. Both of these 24 hour surveillances provide a compensatory diverse method for detecting RCS leakage to compliment the remaining operable systems (i.e., sump level and particulate detection).

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.4

<u>CHANGE NUMBER</u>	<u>JUSTIFICATION</u>
3.4-17	This change extends the surveillance interval for pressurizer heaters from 92 days to 18 months. The operability of the pressurizer heaters enhance the capability of the plant to control RCS pressure and establish natural circulation. The purpose of the surveillance requirement is to detect potential pressurizer heater degradation. This is done by periodically demonstrating that pressurizer heaters are capable of producing power at their design rating by testing the power supply output and by performing electrical checks on heater elements. Heating elements are simple resistive devices which are not prone to complex failure modes. Moreover, heater banks are made up of a number of individually powered heater elements such that a common mode failure of the elements is unlikely. The normal power supply to the heaters is in use during normal power operations and a failure of the power supply would be immediately detectable independent of the heater surveillance. The low failure rate experienced with pressurizer heater elements is indicative that the surveillance interval may be extended without loss in heater reliability. This surveillance interval extension is consistent with the guidance of GL 93-05, "Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation." This proposed change is consistent with Industry Traveler TSTF-93, Rev. 1.
3.4-18	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
3.4-19	CTS LCO 3.4.6.2 states required conditions for the start of a RCP when none have been operating. Another condition for starting an RCP is added that is not specified in NUREG-1431. This added condition provides an <u>OR</u> restraint for the option of a RCP start if the pressurizer water level is less than 50%. The option of pressurizer water level being less than 50% provides plant operational flexibility because it allows a RCP start in the event the primary/secondary temperature difference was not satisfied. The volume in the pressurizer provides space to sustain possible reactor coolant thermal swell without incurring an excessive pressure transient. Consequently the intent to protect the RCS from cold overpressure transients is maintained. The vessel water level method is not present in NUREG-1431 and has been incorporated in the reformatted LCO note. This proposed change meets the intent of NUREG-1431 and is consistent with the DCP licensing basis.
3.4-20	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
3.4-21	DCP design includes three air operated pressurizer PORVs. Two of these PORVs have been designated as "Class I". These two valves provide the reactor vessel low temperature overpressure protection and they provide the means to depressurize the RCS following a SGTR. The identification of Class I has been added to the ITS to make a distinction between these two PORVs that must provide a safety-related function as opposed to the third remaining PORV that is designated as non-Class I. TS 3.4.12 for LTOP applies to the two Class I PORVs but not to the non-Class I PORV.
3.4-22	This change deletes Required Action B.1 consistent with Industry Traveler TSTF-28. The deleted ACTION required that SR 3.4.16.2 be performed within 4 hours. This SR must be performed in order to verify "restoration" of the specific activity to within limits, and does not need to be otherwise required. Further, if the Condition is entered and the plant is in MODE 2 in 4 hours or less, the NUREG Required Action is in conflict with the Note of SR 3.4.16.2 that states that this SR is only required in MODE 1. In addition, this ACTION is an unnecessary burden as the plant is required to be in MODE 3 with $T_{avg} < 500^{\circ}\text{F}$ within 12 hours, exiting the MODE of Applicability.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.4

<u>CHANGE NUMBER</u>	<u>JUSTIFICATION</u>
3.4-23	ITS SR 3.4.12.3 is revised to be consistent with the requirements denoted in LCO 3.4.12 and Industry Traveler WOG-51, Rev. 1. This change clarifies that the SR (to verify the accumulators isolated) is only applicable when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.
3.4-24	Improved SR 3.4.14.1 requires testing of PIVs prior to entering MODE 2 whenever the unit has been in MODE 5 for seven days or longer. This surveillance frequency is not in the DCPD CTS. DCPD tests the valves on an 18 month Frequency which is acceptable based on valve type, interlocks, position indication and system alarm functions. These valves meet the criteria for testing at least every 18 months but do not require additional testing based on when the unit is in cold shutdown. None of these valves change position during normal plant operation. There is little reason to anticipate leakage to initiate in mid cycle when they have not changed position. In conformance with the CTS, DCPD has chosen not to include this surveillance frequency of retesting the valves following an extended period of operation in MODE 5.
3.4-25	Consistent with the CTS, the gross activity limit is added to LCO 3.4.16 Condition B rather than its first reference being in SR 3.4.16.1. This change is also consistent with the treatment of DOSE EQUIVALENT I-131.
3.4-26	DCPD design designates two pressurizer PORVs to provide the function for low temperature overpressure protection. The valves have been designated as a Class I system. The PORVs are air operated. The air supply is not a Class I system, consequently, the design includes a back-up nitrogen Class I system to power the valves in the event the air supply is not available. The surveillance for the nitrogen back-up system is in the DCPD CTS but not present in NUREG-1431. Improved TS SR 3.4.11.3 has been added to provide assurance of the operation of the nitrogen gas supply to the PORVs. Methodology of verifying operability has been moved to the Bases.
3.4-27	TS 3.4.12 Applicability for MODE 6 has an additional qualification if the head closure bolts are not fully de-tensioned. With the bolts fully de-tensioned this TS is not Applicable because the lack of tension provides a pressure relief path across the vessel flange. This Applicability is in the current licensing basis.
3.4-28	This change has added the description of a secured open valve. This description is in use in the DCPD CTS and in plant procedures. This description is also consistent with that used in NUREG-1431 SR 3.5.2.2 and SR 3.5.2.5.
3.4-29	The definition for CHANNEL FUNCTIONAL TEST (CFT) would be retained from the current DCPD TS to the improved TS. CFT is in active use in numerous procedures in the plant. The CFT is used in applications for which the CHANNEL FUNCTIONAL TEST (COT) is not fully suitable. Although CFT and COT definitions appear similar, there is one important difference. Strict adherence to COT requirements includes quantitative adjustments as appropriate to bring setpoints into the desired range. This requirement for quantitative adjustment can not be satisfied in a reasonable manner on some components/sensors/channels due to their design. CFT however, is a qualitative test to determine functionality. A loss of function indicated by the CFT results in a notification to, following existing procedures, restore the functional performance. The CFT is in the current licensing basis.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.4

CHANGE NUMBER

JUSTIFICATION

- 3.4-30 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.4-31 These ACTIONS in ITS 3.4.5 and 3.4.9 are modified to reflect their LCO. The position of the reactor trip breakers and their power supply status of the CRDMs are not LCO requirements; therefore Conditions and ACTIONS are revised. As worded in NUREG-1431, these ACTIONS could preclude certain testing in MODE 3. A more generic ACTION, which assures the rods cannot be withdrawn, replaces the specific method of precluding rod withdrawal. The specific methods are added to the Bases as examples. The revised ACTIONS still assure rod withdrawal is precluded and this detail is not required to be in the TS to provide adequate protection to the public health and safety. No technical changes result from this change. These changes are consistent with Industry Traveler TSTF-87, Rev. 1.
- 3.4-32 In accordance with Industry Traveler TSTF-26, the ACTION would be changed to specify taking the plant to a MODE for which the LCO is not applicable. This change maintains the consistency between the MODE of Applicability and the Required Action which requires the MODE of Applicability to be exited.
- 3.4-33 The Frequency of SR 3.4.2.1 to verify operating RCS loop average temperature at or above [541] °F is changed to 12 hours from the current surveillance Frequency of 30 minutes. The SR to verify operating RCS loop average temperatures every 12 hours is sufficiently frequent to prevent inadvertent violation of the LCO and considers indications and alarms that are continuously available to the operator in the control room. This change is based on Industry Traveler TSTF-27, Rev. 2.
- 3.4-34 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.4-35 This change adds a Note to SR 3.4.11.1 and SR 3.4.11.2 stating that the SRs are only required to be performed in MODES 1 and 2. The ACTIONS Note "LCO 3.0.4 is not applicable" is intended to allow MODE changes for testing purposes (per Bases). This allowance is properly presented as an SR Note. A properly placed exception (i.e., an SR Note exception) would not allow the SR to be considered to be met until the appropriate conditions were available for it to be performed without entering the ACTIONS. The Note to these SRs would allow startup in MODE 3 if the SR had not been performed during the required Frequency, but would limit the exception to prior to entering MODE 2. The change is consistent with Industry Traveler WOG-60.
- 3.4-36 SR 3.4.13.1 and ACTIONS for LCO 3.4.15 are revised with the addition of a Note per Industry Traveler TSTF-116, Rev. 1. The Note addresses the concern that RCS water inventory balance cannot be meaningfully performed unless the unit is operating at or near steady state conditions. The Note added to the surveillance provides an exception for operation at less than steady state conditions. The RCS water inventory balance will only be allowed to be deferred for 12 hours after re-establishing steady state conditions.
- 3.4-37 Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.4

<u>CHANGE NUMBER</u>	<u>JUSTIFICATION</u>
3.4-38	Consistent with Industry Traveler TSTF-105, the details on the method by which the RCS flow rate is verified are moved from SR 3.4.1.4 to the Bases. Moving this information to the Bases allows the use of precision heat balances, elbow taps, and other acceptable methods in order to perform this verification and is consistent with the NUREG-1431 philosophy of moving clarifying information and descriptive details out of the TS to the Bases.
3.4-39	The shutdown requirements of ITS 3.4.11 would require the plant to reduce T_{avg} to $<500^{\circ}\text{F}$ within 12 hours, rather than be in MODE 4, to address the concern of entering [LTOP] LCO 3.4.12 Applicability with inoperable PORVs. For consistency, the shutdown requirements of ITS 3.4.16 are also revised to allow 12 hours to reduce T_{avg} to $<500^{\circ}\text{F}$. This change is consistent with Industry Traveler TSTF-113, Rev. 3.
3.4-40	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
3.4-41	Consistent with the current licensing basis in CTS 3.2.3, LCO 3.4.1 would be revised to reference Tables 3.4.1-1 and 3.4.1-2 for RCS total flow rate limits for Unit 1 and Unit 2 respectively. These tables show allowed reduction in reactor thermal power to account for reduced RCS total flow. DCP LA 60/59 added this allowance to the CTS.
3.4-42	An exception to SR 3.4.14.1 Frequency to leak test PIVs within 24 hours of actuation or flow through the valves has been added for valves 8802A, 8802B and 8703. Testing of these valves under this Frequency is not required as stated in the footnote to current Surveillance 4.4.6.2.1, Table 3.4-1. This change is consistent with the CTS.
3.4-43	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
3.4-44	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
3.4-45	ITS 3.4.12 has been revised to move the Note for Required ACTION B.1 regarding CCP pump swap operations and the Applicability Note for accumulator isolation to the LCO, as discussed in Industry Traveler WOG-51, Rev 1. Plant-specific time allowances for exceeding LCO's number of [Centrifugal Charging] pumps capable of injecting into the RCS are incorporated. [] These Notes detail situations where exceptions to the LCO are permitted and are more appropriately annotated under the LCO.
3.4-46	Consistent with CTS 3/4.1.1.4, "Minimum Temperature for Criticality," ITS LCO 3.4.2 and its Condition A and SR 3.4.2.1 are modified to refer to "operating" RCS loops. Adopting the CTS wording is acceptable since valid T_{avg} measurements are not obtainable for a non-operating loop.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.4

<u>CHANGE NUMBER</u>	<u>JUSTIFICATION</u>
3.4-47	ITS SR 3.4.11.1 contains a Note which exempts the cycling of the block valve when it is closed in accordance with Required Actions of Conditions B or E of LCO 3.4.11. However, Required Action A.1 also directs closure of the block valve when one or more PORVs are inoperable and capable of being manually cycled. The SR Note should also exempt performance when the block valve is closed in accordance with Required Action A.1 as the block valve should not be opened when the PORV is inoperable. This change is consistent with NUREG-1430 and NUREG-1432 in as much as the block valve cycling is exempted under Conditions A, B, and E. Since power to the block valve(s) is maintained in Required Action A.1, the Note to SR 3.4.11.1 will be revised to not require the surveillance performance if the block valve(s) is closed per Condition A. Since power to the block valve(s) is removed in Required ACTIONS B.2 and E.3, the surveillance cannot be met. Given the wording change "met" to "performed" in the Note, the wording of SR 3.4.11.1 is revised to accommodate the Condition B and E exception. This change is consistent with Industry-Traveler WOG-87.
3.4-48	A Note is added to ITS 3.4.8 ACTIONS, indicating that entry into MODE 5, Loops Not Filled, from MODE 5, Loops Filled, is not permitted while LCO 3.4.8 is not met. The addition of this Note is based on the performance of a plant specific LCO 3.0.4 matrix (see CN 01-02-LS1 of the CTS 3/4.0 package).
3.4-49	LCO 3.4.12 [LTOP] System, provides four different methods for pressure relief. Any of the four methods may be used. However, SR 3.4.12.5 requires testing whether or not the equipment is being credited to meet the LCO. The proposed change adds the words "required" to the SR to exempt its performance if the equipment to be tested is not being used to meet the LCO. In addition, two editorial changes were made. The LCO requirement presentation was clarified. Also, the Note to SR 3.4.12.8 was revised to replace "required to be met" with "required to be performed" since the "performed" nomenclature is appropriate here, consistent with the CTS. This change is consistent with Industry Traveler WOG-100.
3.4-50	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
3.4-51	The Note for SR 3.4.1.4 is removed. This is consistent with DCP CTS 4.2.3.5. DCP conducts a measured RCS total flow rate verification on the 18 month frequency in accordance with NUREG-1431.
3.4-52	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(8 Pages)

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.4

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-01	Clarifies intent of wording for the allowance to remove pumps from operation by changing "de-energized" to "removed from operation" consistent with Industry Traveler TSTF-153.	Yes	Yes	Yes	Yes
3.4-02	This change revises Condition A of ITS 3.4.6 to cover any required loop's inoperability and adds Required ACTION A.2 indicating that cooldown to MODE 5 is only required if an RHR loop is OPERABLE. Condition B is deleted.	Yes	Yes	Yes	Yes
3.4-03	The current TS allows 1 hour for all RHR pumps to be removed from operation [].	Yes	Yes	Yes	Yes
3.4-04	The symbol ">" in DCPD LCO 3.4.8, NOTES 1.a., is replaced with the words "at least" to make this LCO Note wording consistent with LCOs 3.4.6 and 3.4.7 Note wording.	Yes	No	No	No
3.4-05	This change is being made consistent with the current assumptions used in the analysis. The analysis credits three operational restrictions below 350°F, to ensure that the reactor vessel is protected.	No, these operational restrictions are not in the CTS.	Yes	No, similar analysis not contained in CTS.	No, similar analysis not contained in CTS.
3.4-06	Plant specific safety analyses do not allow injection from safety injection pumps but do allow centrifugal charging pumps (CCPs) injection.	Yes, DCPD analysis assumes one CCP only.	Yes, CPSES analysis assumes two CCPs.	Yes, WCNOG analysis assumes one CCP only.	Yes, Callaway analysis assumes one CCP only.
3.4-07	The word "all" in the DCPD LCO 3.4.12 LTOP Applicability is replaced by the word "any" to make the Applicability wording consistent with the Bases.	Yes	No	No	No
3.4-08	The current licensing basis as contained in the Technical Specifications requires performance of this surveillance on a frequency of 72 hours.	No, DCPD LTOP design does not use RHR relief valves.	Yes	Yes, see Amendment No. 49.	Yes, see OL Amendment No. 42.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.4

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-09	The plant does not have manual RHR suction isolation valves. The motor-operated suction isolation valves (2 per relief valve line) are surveilled in accordance with SR 3.4.12.4.	No, DCPD LTOP design does not use RHR relief valves.	Yes	Yes	Yes
3.4-10	The DCPD plant-specific limiting temperature specified in degrees, below which the RCS must not be subject to low temperature overpressure is replaced by the generic statement "the temperature below which LTOP is required as specified in the PTLR."	Yes	No	No	No
3.4-11	The plant does not have the RHR autoclosure portion of the RHR System interlock as the system was deleted from the design. However, the portion of the interlock which prevents the valves from opening when system pressure is in excess of the setpoint has been retained.	No, the valve interlock is not in the CTS.	Yes	Yes, see Amendment No. 49	Yes, see OL Amendment No. 42.
3.4-12	In conformance with the CTS, the RHR Isolation Valves which are RCS PIVs are excluded from being retested following an extended period of operation in MODE 5.	No, RHR valve testing after MODE 5 operation is not in the CTS.	Yes	No, WCNOG does not have this exclusion.	No, Callaway does not have this exclusion.
3.4-13	The RHR Isolation Valves which are RCS PIVs are excluded from being retested following flow through the valves.	No, not in CTS.	Yes	Yes	Yes
3.4-14	This change revises the grouping of required detectors to be consistent with the CTS and with Reg. Guide 1.45 Regulatory Position C.3.	Yes	Yes	Yes	Yes
3.4-15	LCO 3.0.4 is applied to Condition [D] (required containment atmosphere gaseous radioactivity monitor and containment cooler condensate monitoring system inoperable).	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.4

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-16	Consistent with CTS requirements [and with Condition B,] a requirement to perform 24 hour containment atmosphere samples is added when a gaseous monitor is inoperable. [] Also, consistent with Conditions A and B, the performance of SR 3.4.13.1 every 24 hours as an alternative to analyzing a containment atmosphere sample is allowed.	Yes	Yes	Yes	Yes
3.4-17	This change revises the pressurizer heater capacity surveillance requirement frequency from 92 days to 18 months.	Yes	Yes	Yes	Yes, part of CTS per Amendment 105.
3.4-18	The Applicability for ITS LCOs 3.4.10 and 3.4.11, LCO 3.4.12 and LCO Note 2 for ITS 3.4.5, are modified to be consistent with current [LTOP] analyses requirements.	No, DCPD has different LTOP requirements.	No, 2 CCPs are allowed per CTS.	Yes	Yes
3.4-19	For DCPD, this change adds an LCO condition to allow RCP start with pressurizer water level less than 50%, per CTS.	Yes	No, not in CTS.	No, not in CTS.	No, not in CTS.
3.4-20	This change adds a provision to allow one or more SI pumps to be capable of injecting into RCS in MODES 5 and 6 when RCS water level is below the top of the reactor vessel flange for the purpose of protecting the decay heat removal function, per CTS.	No, not part of CTS.	No, not part of CTS.	Yes	Yes
3.4-21	Two DCPD pressurizer PORVs have been identified as Class I to make the distinction between these and the third PORV that is designated non-Class I.	Yes	No	No	No
3.4-22	This change deletes ISTS 3.4.16 ACTION B.1 per Industry Traveler TSTF-28.	Yes	Yes	Yes	Yes
3.4-23	ITS SR 3.4.12.3 is revised to be consistent with the requirements denoted in LCO 3.4.12.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.4

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-24	Improved SR 3.4.14.1 requires testing of PIVs prior to entering MODE 2 whenever the unit has been in MODE 5 for seven days or longer. This surveillance Frequency is not in the DCPD CTS. In conformance with the CTS, DCPD has chosen not to include this surveillance Frequency.	Yes	No	No	No
3.4-25	The gross specific activity limit is added to LCO 3.4.16, Condition B, rather than having its first reference occurring in SR 3.4.16.1.	Yes	Yes	Yes	Yes
3.4-26	This change adds surveillance for PORV safety related nitrogen supply, per DCPD CTS.	Yes	No	No	No
3.4-27	Applicability for LCO 3.4.12, MODE 6, is revised to include an additional qualification if the head closure bolts are not fully de-tensioned, per DCPD CTS.	Yes	No	No	No
3.4-28	This change adds a DCPD-specific description of a secured open valve.	Yes	No	No	No
3.4-29	The use of CHANNEL FUNCTIONAL TEST (CFT) would be retained from the current DCPD TS to the improved TS.	Yes	No	No	No
3.4-30	An LCO 3.0.4 exception is added to the ACTIONS LCO 3.4.12.	No, not in CTS	Yes	Yes	Yes
3.4-31	Condition C and Required ACTION D.1 of ITS 3.4.5 and Condition A of ITS 3.4.9 are modified to reflect generic wording to assure that the rods are fully inserted and cannot be withdrawn. This change is consistent with Industry Traveler TSTF-87, Rev. 1.	Yes	Yes	Yes	Yes
3.4-32	In accordance with Industry Traveler TSTF-26, 3.4.2 ACTION would be changed to specify taking the plant to a MODE for which the LCO is not applicable.	Yes	Yes	Yes	Yes

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-33	The Frequency of SR 3.4.2.1 is changed to "12 hours". This change is based on Industry Traveler TSTF-27, Rev. 2.	Yes	Yes	Yes	Yes
3.4-34	Retains CPSES current TS which requires that the precision RCS flow measurement be performed prior to exceeding 85% RTP.	No	Yes	No	No
3.4-35	Adds a Note to SR 3.4.11.1 and SR 3.4.11.2 stating that the SRs are only required to be performed in MODES 1 and 2, per Industry Traveler WOG-60.	Yes	Yes	Yes	Yes
3.4-36	SR 3.4.13.1 and LCO 3.4.15 ACTIONS are revised with the addition of a Note requiring steady state conditions per Industry Traveler TSTF-116, Rev. 1.	Yes	Yes	Yes	Yes
3.4-37	The primary to secondary leakage limits are revised per Callaway OL Amendment No. 116 dated October 1, 1996.	No	No	No	Yes
3.4-38	Consistent with Industry Traveler TSTF-105, the details on the method by which the RCS flow rate is verified are moved from SR 3.4.1.4 to the Bases.	Yes	Yes	Yes	Yes
3.4-39	The shutdown requirements of ITS 3.4.11 would require the plant to reduce T_{avg} to <500°F within 12 hours, rather than MODE 4, to address the concern of entering [LTOP] LCO 3.4.12 Applicability with inoperable PORVs. For consistency, the shutdown requirements of ITS 3.4.16 are also revised to allow 12 hours to reduce T_{avg} to <500°F. This change is consistent with Industry Traveler TSTF-113, Rev. 3.	Yes	Yes	Yes	Yes
3.4-40	Consistent with Industry Traveler WOG-99, the Note to SR 3.4.1.4 would be modified to specify a plant-specific power and to provide additional time to perform an RCS precision flow rate measurement.	No, see CN 3.4-51.	No, see CN 3.4-34.	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.4

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-41	LCO 3.4.1 is revised to reference Tables 3.4.1-1 and 3.4.1-2 for RCS total flow rate limits for DCPD Unit 1 and Unit 2 respectively.	Yes, allowance added per Amendments 60/59	No	No	No
3.4-42	An exception to SR 3.4.14.1 Frequency to leak test PIVs 8802A, 8802B and 8703 has been added. This change is consistent with the DCPD CTS.	Yes	No	No	No
3.4-43	A new Condition C is added to LCO 3.4.1 to reflect the CTS of Wolf Creek for RCS flow rate.	No	No	Yes	No
3.4-44	Steam generator levels for MODES 3, 4, and 5 are specified to ensure SG tubes are covered. The Callaway CTS did not ensure tube coverage.	No	No	No	Yes
3.4-45	ITS 3.4.12 has been revised to move the Note for Required ACTION B.1 regarding CCP pump swap operations and the Applicability Note for accumulator isolation to the LCO, as discussed in Industry Traveler WOG-51, Rev 1. Plant-specific time allowances for exceeding LCO's number of [RHR] pumps capable of injecting into the RCS are incorporated, [].	Yes	No, operation of 2 CCPs are allowed per CTS; also see CN 3.4-52.	Yes	Yes
3.4-46	ITS LCO 3.4.2 and its Condition A and SR 3.4.2.1 are modified to refer to "operating" RCS loops.	Yes	Yes	Yes	Yes

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-47	ITS SR 3.4.11.1 contains a Note which exempts the cycling of the block valve when it is closed in accordance with Required Actions of Condition B or E of LCO 3.4.11. However, Required ACTION A.1 also directs closure of the block valve when one or more PORVs are inoperable and capable of being manually cycled. The SR Note should also exempt performance when the block valve is closed in accordance with Required ACTION A.1 as the block valve should not be opened when the PORV is inoperable, per Industry Traveler WOG-87.	Yes	Yes	Yes	Yes
3.4-48	A Note is added to ITS 3.4.8 ACTIONS, indicating that entry into MODE 5, Loops Not Filled, from MODE 5, Loops Filled, is not permitted while LCO 3.4.8 is not met.	Yes	Yes	Yes	Yes
3.4-49	This change reorganizes the presentation of ITS LCO 3.4.12, adds the word "required" to ITS SR 3.4.12.5, and changes the word "met" to "performed" in ITS SR 3.4.12.8, per Industry Traveler WOG-100.	Yes	Yes	Yes	Yes
3.4-50	This change is consistent with CTS SR 4.4.9.3.3. The 12 hour Frequency applies to vent pathway(s) that are "not locked, sealed, or otherwise secured in the open position". The wording added to ITS SR 3.4.12.5 is also consistent with the format used in similar ITS 3.6 SRs. The 31 day Frequency is also revised to be consistent with CTS SR 4.4.9.3.3.	No, adopting ITS format.	No, adopting ITS format.	Yes	Yes
3.4-51	The Note for SR 3.4.1.4 is removed. This is consistent with DCPD CTS 4.2.3.5. DCPD conducts a measured RCS total flow rate verification on the 18 month frequency in accordance with NUREG-1431.	Yes	No	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.4

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.4-52	Consistent with Industry Traveler WOG-51, Rev 1, the Note concerning accumulator isolation is moved from the Applicability to the LCO.	No, see CN 3.4-45.	Yes	No, see CN 3.4-45.	No, see CN 3.4-45.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-115	Not used.	N/A	N/A	N/A	N/A
3.3-116	ACTION J of ITS 3.3.2 is not used since DCPD does not rely on motor-driven AFW pump start with loss of both main FW pumps.	Yes	No	No	No
3.3-117	This change to ITS 3.3.1 Condition R reflects CTS Table [3.3-1, ACTION Statement 12] which was based on NRC Generic Letter 85-09.	Yes	No, not in CTS.	Yes	Yes
3.3-118	This change is for consistency with ITS 3.7.10 Condition [G].	Yes	Yes	Yes	Yes
3.3-119	This change reflects Callaway-specific BDMS analysis restrictions associated with RCS mixing volume and dilution flow rate. These are administratively controlled under the CTS, as approved in OL Amendment No. 94 dated March 7, 1995. However, with the conversion to ITS 3.3.9, these analysis assumptions should be included in the body of the TS.	No	No	No	Yes
3.3-120	ITS 3.3.1 Condition D is revised to reflect ITS SR 3.2.4.2 and CN 3.2-15 in the 3/4.2 package.	Yes	Yes	Yes	Yes
3.3-121	For Callaway, ITS 3.3.9 is revised to reflect that only one BDMS train is required OPERABLE in MODE 5 and that the suspension of positive reactivity additions and accelerated SDM verifications are required only if no source range neutron flux indicator is OPERABLE.	No	No	No	Yes
3.3-122	ITS 3.3.1 APPLICABILITY Note (b) for Functions 1, 5, 19-21 and Conditions C and K are revised to replace ACTIONS requiring the RTBs to be opened with ACTIONS that ensure subcriticality is maintained (i.e., by fully inserting all rods and ensuring the Rod Control System is incapable of rod withdrawal) yet do not initiate a feedwater isolation (P-4 and low T_{avg}) in MODE 3, consistent with Traveler TSTF-135.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-123	This change deletes ACTION L.2 and renumbers L.3 since the requirement to close the unborated water source valves is not in the CTS.	Yes; refer to LA 28/27.	No, see CN 3.3-41.	No, see CN 3.3-41.	No, see CN 3.3-41.
3.3-124	Consistent with the CTS Table 4.3-1, Note [15], the notes for ITS SR 3.3.1.4 and Table 3.3.1-1, Function 20 are modified to clarify that the SR is required for the reactor trip bypass breaker local manual shunt trip only. The Bases for SR 3.3.1.14 clearly state that SR 3.3.1.14 includes the automatic undervoltage trip of the reactor trip bypass breakers. The Note (k) added to Table 3.3.1-1, Function 20 clarifies the Applicability of the undervoltage and shunt trip mechanisms to include those functions of the reactor trip bypass breakers when in use.	Yes	Yes	Yes	Yes
3.3-125	ITS SR 3.3.1.11 is modified by a Note that requires verification that the time constants are adjusted to the prescribed values. The addition of this Note is consistent with SR 3.3.1.10 and is required because SR 3.3.1.11 is used for the Power Range Neutron Flux - High Positive Rate [and High Negative Rate] trip functions which have time constants associated with their calibration.	Yes	Yes	Yes	Yes
3.3-126	The surveillance frequency associated with the RWST Level - Low-Low ESFAS function is restricted to 9 months by the drift allowance in the CPSES-specific setpoint study. Therefore, a new surveillance requirement (SR 3.3.2.12) is specified that replaces SR 3.3.2.9 used for those ESFAS functions with 18 month surveillance frequencies. This modification is consistent with the CPSES CTS.	No, not in CTS.	Yes	No, not in CTS.	No, not in CTS.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-127	The MODE 2 applicability for the undervoltage RCP start of the steam-driven AFW pump is deleted and the surveillance frequency is revised per the DCPD CTS. Thus, the Required Actions of ACTION I are revised to include entering MODE 2 for function 6.g and MODE 3 for function 5.b, and the required surveillance is changed from SR 3.3.2.7 to SR 3.3.2.8.	Yes	No	No	No
3.3-128	This change revises ITS Table 3.3.4-1 to be consistent with DCPD CTS 3.3.3.5.	Yes	No	No	No
3.3-129	Consistent with the CPSES CTS, the ITS requirement to have the Loss of Power Diesel Generator Start Instrumentation (ITS 3.3.5) applicable in Modes 5 and 6 when the associated DG is required to OPERABLE by ITS LCO 3.8.2 "AC Sources - Shutdown" is deleted. In these modes, the Reactor Coolant System or Reactor Vessel cavity temperatures are low enough that the time available for the reactor operators to manually start the diesel generators is adequate. Thus, there is no need to require the automatic loss of power DG start instrumentation to be operable in these modes.	No	Yes	No	No
3.3-130	Consistent with the CTS and the CPSES design, the Conditions, Required Actions and Surveillances are not applicable to the 6.9 kV Preferred Offsite Source Undervoltage function if the associated source breaker is open. When the associated source breaker is open, credit is not being taken for the immediate availability of the 6.9 kV Preferred Offsite AC power source.	No	Yes	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.3

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-131	ITS 3.3.5 Condition B is replaced with new Conditions B, C, D, and E. Condition C in the ISTS is changed to Condition F. The CPSES CTS have specific actions for the various bus undervoltage and degraded voltage function. These actions allow an appropriate amount of time to restore an inoperable channel or declare the associated power source or bus inoperable and take action to isolate an inoperable power source. These actions are a proper way to respond to the inoperable channels because the actions result in taking the Required Actions in ITS 3.8 associated with the affected power source or bus. The new Conditions match the Actions of the CTS.	No	Yes	No	No
3.3-132	The trip setpoints for the loss of power diesel generator start instrumentation are relocated to a licensee controlled document. This approach is consistent with a format allowed by a reviewer's note for the RTS and ESFAS instrumentation.	No -adopted ITS format.	Yes	No, adopted ITS format.	No, adopted ITS format.
3.3-133	This change revises ITS LCO 3.3.5 and SR 3.3.5.3 to include the DG start sequence delay timers from DCPP CTS Table 3.3-4.	Yes	No	No	No
3.3-134	This change is Wolf Creek specific to revise the NOTE in Condition K of ITS 3.3.2 consistent with CTS Table 3.3-3 Action 16 for Function 7b and Amendment 43 to provide 4 hours foran additional channel to be placed in bypass for surveillance testing of other channels.	No	No	Yes	No
3.3-135	A MODE change restriction has been added per the matrix discussed in CN 1-02-LS-1 of the ITS 3.0 package.	Yes	Yes	Yes	Yes
3.3-136	The TADOT perfomed under ITS SR 3.3.2.7 includes verification of relay setpoints since the trip actuating devices being tested are the same circuits tested under ITS SR 3.3.5.2.	No, adopted ISTS format.	No, adopted ISTS format.	Yes	Yes
{3.3-137	The Condition for Function 4.c is changed from Condition D to E consistant with the DCPP CTS.	Yes	No	No	No}

JLS Conversion to Improved Technical Specifications

Diablo Canyon Power Plant

Docket # 50-275

Accession # 9706230042

Date 6/2/97 of Ltr

Regulatory Docket File

CTS 3/4.5 - Emergency Core Cooling System

ITS 3.5 - Emergency Core Cooling System



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.5

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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

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CROSS-REFERENCE TABLE FOR 3/4.5
Sorted by Current TS

Current TS				Improved TS			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.5.1	LCO			3.5.1	LCO		
3.5.1	LCO	a.		3.5.1.1	SR		
3.5.1	LCO	b.	01-08-A	3.5.1.2	SR		
3.5.1	LCO	c.		3.5.1.4	SR		
3.5.1	LCO	d.		3.5.1.3	SR		
3.5.1	APP			3.5.1	APP		
3.5.1	APP	* Note	01-03-A	3.5.1	APP		3.5-1
3.5.1	ACTION	a.	01-04-LS8	3.5.1	ACTIONS	A, C	
3.5.1	ACTION	a.	01-06-A	3.5.1	ACTIONS	D	
3.5.1	ACTION	a.	01-03-A	3.5.1	ACTIONS	C.2	
3.5.1	ACTION	b.	01-05-LS9	3.5.1	ACTIONS	B, C	
3.5.1	ACTION	b.	01-06-A	3.5.1	ACTIONS	D	
3.5.1	ACTION	b.	01-03-A	3.5.1	ACTIONS	C.2	
4.5.1.1	SR	a. 1)		3.5.1.2	SR		
4.5.1.1	SR	a. 1)		3.5.1.3	SR		
4.5.1.1	SR	a. 2)		3.5.1.3	SR		
4.5.1.1	SR	b.	01-07-LG 01-08-A	3.5.1.4	SR		
4.5.1.1	SR	c.		3.5.1.5	SR		
3.5.2	LCO		02-01-LG	3.5.2	LCO		
3.5.2	LCO	a.	02-01-LG			Not Used	
3.5.2	LCO	b.	02-01-LG			Not Used	
3.5.2	LCO	c.	02-01-LG			Not Used	
3.5.2	LCO	d.	02-01-LG			Not Used	
3.5.2	LCO	e.	02-01-LG			Not Used	

CROSS-REFERENCE TABLE FOR 3/4.5
Sorted by Current TS

Current TS				Improved TS			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.5.2	APP		02-02-LS1	3.5.2	APP		
3.5.2	ACTION	a.	02-01-LG	3.5.2	ACTIONS	A	
3.5.2	ACTION	a.	02-03-LS2	3.5.2	ACTIONS	A	
3.5.2	ACTION	a.* Note		3.5.2	ACTIONS	A Note	3.5-7
3.5.2	ACTION	b.	02-04-TR2			Not Used	
4.5.2	SR	a.	02-07-A	3.5.2.1	SR		
4.5.2	SR	b. 1)	02-16-LG	3.5.2.3	SR		
4.5.2	SR	b. 2)		3.5.2.2	SR		
4.5.2	SR	c. 1)	02-09-LG			Not Used	
4.5.2	SR	c. 2)	02-09-LG			Not Used	
4.5.2	SR	d.		3.5.2.8	SR		
4.5.2	SR	e. 1)	02-11-TR1	3.5.2.5	SR		
4.5.2	SR	e. 2)	02-11-TR1	3.5.2.6	SR		
4.5.2	SR	f.	02-12-LG	3.5.2.4	SR		
4.5.2	SR	f. 1)	02-12-LG			Not Used	
4.5.2	SR	f. 2)	02-12-LG			Not Used	
4.5.2	SR	f. 3)	2-12-LG			Not Used	
4.5.2	SR	g. 1)	02-13-A			Not Used	
4.5.2	SR	g. 2)		3.5.2.7	SR		3.5-3
4.5.2	SR	h.1) a)	02-15-LG			Not Used	
4.5.2	SR	h. 1) b)	02-15-LG			Not Used	
4.5.2	SR	h. 1) c)	02-15-LG			Not Used	
4.5.2	SR	h. 1) d)	02-15-LG			Not Used	
4.5.2	SR	h. 2) a)	02-15-LG			Not Used	
4.5.2	SR	h. 2) b)	02-15-LG			Not Used	
4.5.2	SR	h. 2) c)	02-15-LG			Not Used	

CROSS-REFERENCE TABLE FOR 3/4.5
Sorted by Current TS

Current TS				Improved TS			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
4.5.2	SR	h. 2) d)	02-15-LG			Not Used	
4.5.2	SR	i.	02-15-LG			Not Used	
3.5.3	LCO		03-01-LG	3.5.3	LCO		
3.5.3	LCO	a. * Note	03-02-LS4	3.4.12	APP	NOTE 1	3.4-01
3.5.3	LCO	a. * Note	03-11-LG	3.4.12	APP		3.4-10
3.5.3	LCO	b.	03-01-LG			Not Used	
3.5.3	LCO	c.	03-01-LG			Not Used	
3.5.3	LCO	d.	03-01-LG			Not Used	
3.5.3	APP			3.5.3	APP		
3.5.3	ACTION	a.	03-03-LS5	3.5.3	ACTIONS	B, C	
3.5.3	ACTION	b.	03-04-LG	3.5.3	ACTIONS	A	
3.5.3	ACTION	c.	03-05-TR2			Not Used	
4.5.3.1	SR			3.5.3	LCO		
4.5.3.1	SR	# Note	03-06-A	3.5.3.1	LCO	Note	3.5-6
4.5.3.1	SR		03-10-LS6	3.5.3.1	SR		
4.5.3.2	SR		03-02-LS4	3.4.12	LCO, SR		
4.5.3.2	SR		03-11-LG	3.4.12	APP		
4.5.3.2	SR	* Note		3.4.12	APP	Note 2	3.4-01
4.5.3.2	SR		03-08-LG			Not Used	
3.5.5	LCO	a.	05-03-A	3.5.4.2	SR		
3.5.5	LCO	b.		3.5.4.3	SR		
3.5.5	LCO	c.		3.5.4.1	SR		
3.5.5	APP			3.5.4	APP		
3.5.5	ACTION	a.	05-04 LS12	3.5.4	ACTIONS	A, C	

CROSS-REFERENCE TABLE FOR 3/4.5
Sorted by Current TS

Current TS				Improved TS			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.5.5	ACTION	b.	05-04-LS12	3.5.4	ACTIONS	B, C	
4.5.5	SR	a. 1)		3.5.4.2	SR		
4.5.5	SR	a. 2)		3.5.4.3	SR		
4.5.5	SR	b.		3.5.4.1	SR		

CROSS-REFERENCE TABLE FOR 3.5
Sorted by Improved TS

Current TS				Improved TS			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.5.1	LCO		01-07-A	3.5.1	LCO		
3.5.1	APP	*NOTE	01-03-A	3.5.1	APP		3.5-1
3.5.1	ACTION	a.	01-04-LS8	3.5.1	ACTIONS	A, C	
3.5.1	ACTION	b.	01-05-LS9	3.5.1	ACTIONS	B, C	
3.5.1	ACTION	a.			ACTIONS	C.1	
3.5.1	ACTION	b.			ACTIONS	C.1	
3.5.1	ACTION	a.	01-03-A	3.5.1	ACTIONS	C.2	3.5-1
3.5.1	ACTION	b.	01-03-A		ACTIONS	C.2	3.5.1
3.5.1	Not Used			3.5.1	ACTIONS	D	
4.5.1.1	SR	a. 2)		3.5.1.1	SR		
4.5.1.1	SR	a. 1)		3.5.1.2	SR		
4.5.1.1	Not Used			3.5.1.3	SR		
4.5.1.1	SR	b.	01-07-LG 01-08-A	3.5.1.4	SR		
4.5.1.1	SR	c.		3.5.1.5	SR		3.5-1
3.5.2	LCO		02-01-LG	3.5.2	LCO		
3.5.2	APP		02-02-LS1	3.5.2	APP		
3.5.2	ACTION	a. *NOTE	LAR 96-03	3.5.2	ACTIONS	A NOTE	3.5-7
3.5.2	ACTION	a.		3.5.2	ACTIONS	B.1, B.2	
4.5.2	SR	a.	02-07-A	3.5.2.1	SR		
4.5.2	SR	b. 2)		3.5.2.2	SR		
4.5.2	SR	b. 1)	02-16-LG	3.5.2.3	SR		
4.5.2	SR	f., f 1), 2), 3)	02-12-LG	3.5.2.4	SR		
4.5.2	SR	e. 1)	02-11-TR1	3.5.2.5	SR		
4.5.2	SR	e. 2)	02-11-TR1	3.5.2.6	SR		

CROSS-REFERENCE TABLE FOR 3/4.5
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
4.5.2	SR	g. 2)		3.5.2.7	SR		3.5-3
4.5.2	SR	d.		3.5.2.8	SR		
3.5.3	LCO		03-01-LG	3.5.3	LCO		
3.5.3	LCO	Not Used		3.5.3	LCO	Note	3.5-6
3.5.3	APP			3.5.3	APP		
3.5.3.	ACTION	b.	03-04-LG	3.5.3	ACTIONS	A	
3.5.3.	ACTION	a.	03-03-LS5	3.5.3	ACTIONS	B, C	
4.5.3.1	SR	# Note	03-06-A	3.5.3.1	SR	Note	3.5-6
4.5.3.1	SR		03-10-LS6	3.5.3.1	SR		
3.5.5	LCO	a., b., c.		3.5.4	LCO		
3.5.5	APP			3.5.4	APP		
3.5.5	ACTION	a.	05-04-LS12	3.5.4	ACTIONS	A, C	
3.5.5	ACTION	b.	05-04-LS12	3.5.4	ACTIONS	B, C	
4.5.5	SR	b.		3.5.4.1	SR	Note	
4.5.5	SR	b.		3.5.4.1	SR		
4.5.5	LCO	a. 1)		3.5.4.2	SR		
4.5.5	SR	a. 2)		3.5.4.3	SR		
3.4.6.2	LCO	e.	06-06-A	3.5.5	LCO		3.5-5
3.4.6.2	APP		06-08-LS9	3.5.5	APP		
3.4.6.2	ACTION	b.	06-09-LS10	3.5.5	ACTIONS	A	3.5-5
3.4.6.2	ACTION	b.		3.5.5	ACTIONS	B.1, B.2	
4.4.6.2.1	SR	a.	06-14-A	3.5.5.1	SR		3.5-5

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

Methodology for Cross-Reference Tables
(Continued)

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).
Note:	The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

**Methodology for Cross-Reference Tables
(Continued)**

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
3.5.1.....	3/4 5-1
3.5.2.....	3/4 5-3
3.5.3.....	3/4 5-7
3.5.4.....	N/A
3.5.5.....	3/4 5-11

Methodology (2 Pages)

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant accumulator shall be OPERABLE with:

- a. The isolation valve open and power removed.
- b. A contained borated water volume of between 836 and 864 cubic feet ~~±50.8% and ±72.6%~~ of borated water. 01-08-A
- c. A boron concentration of between 2200 and 2500 ppm, and
- d. A nitrogen cover-pressure of between 595.5 and 647.5 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

- a. With ~~three accumulators OPERABLE and with one accumulator inoperable - except as a result of a closed isolation valve, due to boron concentration not within limits - restore the inoperable accumulator to OPERABLE status within 1 hour - 72 hours - or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN reduce RCS pressure to ≤1000 psig within the following 6 hours.~~ 01-06-A
01-04-LS8
01-03-A
- b. With ~~three accumulators OPERABLE and one accumulator inoperable due to the isolation valve being closed, for reasons other than boron concentration not within limits, either immediately open the isolation valve or restore accumulator to OPERABLE status within 1 hour or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours, and reduce pressurizer RCS pressure to 1000 psig within the following 6 hours.~~ 01-06-A
01-05-LS9
01-03-A

*Pressurizer RCS pressure above 1000 psig. 01-03-A

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

- a. At least once per 12 hours by:
 - 1) Verifying that the contained b^{or}ated water volume and nitrogen cover-pressure in the tanks are within their limits, and
 - 2) Verifying that each accumulator isolation valve is open ~~and power is removed.~~

- b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to ~~1% of tank volume~~ ~~5.6%~~ ~~of narrow range indicated level~~ by verifying the boron concentration of the accumulator solution. This surveillance is not required when the volume increase makeup source is the RWST and the RWST has not been diluted since verifying that the RWST boron concentration is equal to or greater than the accumulator boron concentration limit. and: 01-08-A

- c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is disconnected by sealing the breaker in the open position. 01-07-LG

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two Emergency Core Cooling System (ECCS) ~~trains~~ subsystems shall be OPERABLE. ~~with each subsystem comprised of:~~

02-01-LG

- a. ~~One OPERABLE centrifugal charging pump.~~
- b. ~~One OPERABLE Safety Injection pump.~~
- c. ~~One OPERABLE Residual Heat Removal heat exchanger.~~
- d. ~~One OPERABLE Residual Heat Removal pump, and~~
- e. ~~An OPERABLE flow path capable of taking suction from the Refueling Water Storage Tank on a Safety Injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.~~

APPLICABILITY: MODES 1, 2, and 3. #

02-02-LS1

ACTION:

a. ~~With one or more ECCS trains subsystem inoperable, and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, restore the inoperable subsystem train(s) to OPERABLE status within 72* hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.~~

02-01-LG

02-03-LS2

b. ~~In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~

02-04-TR2

~~In MODE 3, both SI injection pump flow paths may be isolated by closing the isolation valve(s) for up to 2 hours to perform pressure isolation valve testing per SR ([4.4.6.2.2.1])~~

02-02-LS1

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ~~two~~ ECCS subsystem ~~trains~~ shall be demonstrated OPERABLE:

- a. At least once each 12 hours by verifying that the following valves are in the indicated ~~listed~~ positions with power to the valve operators removed:

02-07-A

<u>Valve Number</u>	<u>Valve Function</u>	<u>Valve Position</u>
8703	RHR to RCS Hot Legs	Closed
8802A	Safety Injection to RCS Hot Legs	Closed
8802B	Safety Injection to RCS Hot Legs	Closed
8809A	RHR to RCS Cold Legs	Open
8809B	RHR to RCS Cold Legs	Open
8835	Safety Injection to RCS Cold Legs	Open
8974A	Safety Injection Pump Recir. to RWST	Open
8974B	Safety Injection Pump Recir. to RWST	Open
8976	RWST to Safety Injection Pumps	Open
8980	RWST to RHR Pumps	Open
8982A	Containment Sump to RHR	Closed
8982B	Containment Sump to RHR	Closed
8992	Spray Additive Tank to Eductor	Open
8701	RHR Suction	Closed
8702	RHR Suction	Closed

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

- b. At least once per 31 days by:
- 1) Verifying that the ECCS piping is full of water by ~~venting the ECCS pump casings and accessible discharge piping high points, and~~ 02-16-LG
 - 2) Verifying that each ~~ECCS~~ valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.
- c. ~~By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suction during LOCA conditions. This visual inspection shall be performed:~~ 02-09-LG
- ~~1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and~~
 - ~~2) At least once daily of the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.~~
- d. At least once per 18 months by: a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion:
- e. At least once per 18 months by:
- 1) Verifying that each automatic valve in the flow path ~~that is not locked, sealed, or otherwise secured in position, actuates to its correct position on an actual or simulated Safety Injection actuation test signal.~~ 02-17-A
02-11-TR1
 - 2) Verifying that each of the following ~~ECCS~~ pumps starts automatically upon receipt of a ~~Safety Injection an actual or simulated actuation test signal.~~ 02-11-TR1
 - a) Centrifugal charging pump.
 - b) Safety Injection pump, and
 - c) Residual Heat Removal pump.
- f. ~~By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:~~ 02-12-LG

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

- 1) ~~Centrifugal charging pump \geq 2400 psid.~~
- 2) ~~Safety Injection pump \geq 1455 psid. and~~
- 3) ~~Residual Heat Removal pump \geq 165 psid.~~

~~Verifying, in accordance with the Inservice Testing Program, that each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.~~

02-12-LG

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

- 1) ~~Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and~~
- 2) At least once per 18 months.

02-13-TR3

Charging Injection
Throttle Valves

Safety Injection
Throttle Valves

8810A
8810B
8810C
8810D

8822A
8822B
8822C
8822D

~~h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:~~

02-15-LG

- 1) ~~For centrifugal charging pumps, with a single pump running:~~
 - a) ~~The sum of injection line flow rates, excluding the highest flow rate, is greater than or equal to 299 gpm, and~~
 - b) ~~The total flow rate through all four injection lines is less than or equal to 461 gpm, and~~
 - e) ~~The difference between the maximum and minimum injection line flow rates is less than or equal to 15.5 gpm, and~~

SURVEILLANCE REQUIREMENTS (continued)

- d) ~~The total pump flow rate is less than or equal to 560 gpm.~~
- 2) ~~For safety injection pumps, with a single pump running:~~
 - a) ~~The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 427 gpm, and~~
 - b) ~~The total flow through all four injection lines is less than or equal to 650 gpm, and~~
 - c) ~~The difference between the maximum and minimum injection line flow rates is less than or equal to 20.0 gpm, and~~
 - d) ~~The total pump flow rate is less than or equal to 675 gpm.~~
- i. ~~By performing a flow test, during shutdown, following completion of modifications to the RHR system that alter the system flow characteristics, and verifying that with a single pump running, and delivering to all four cold legs, a total flow rate greater than or equal to 3976 gpm.~~

02-15-LG

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ~~One~~ ECCS subsystem comprised of the following ~~train~~ shall be OPERABLE.*#

~~03-01-LG~~

~~03-06-A~~

~~03-02-LS4~~

- a. ~~One OPERABLE centrifugal charging pump.*~~
- b. ~~One OPERABLE Residual Heat Removal heat exchanger.~~
- c. ~~One OPERABLE Residual Heat Removal pump, and~~
- d. ~~An OPERABLE flow path capable of taking suction from the Refueling Water Storage Tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.~~

APPLICABILITY: MODE 4.

ACTION:

- a. ~~With no the required ECCS centrifugal charging pump (CCP) subsystem OPERABLE inoperable because of the inoperability of either the centrifugal charging pump or the flow path from the Refueling Water Storage Tank, restore at least one required ECCS CCP subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 24 hours.~~
- b. ~~With no the required ECCS Residual Heat Removal (RHR) subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, inoperable, restore at least one required ECCS RHR subsystem to OPERABLE status immediately, or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.~~
- c. ~~In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~

~~03-03-LS5~~

~~03-04-LG~~

~~03-05-TR2~~

*A maximum of one centrifugal charging pump shall be OPERABLE capable of injecting into the RCS whenever the temperature of one or more of the RCS cold legs is less than or equal to 270°F, below the temperature where LTOP is required

~~03-02-LS4~~

~~03-11-LG~~

EMERGENCY CORE COOLING SYSTEMS

ACTION (continued)

as specified in the Pressure Temperature Limits Report.

~~#An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation.~~

03-06-A

SURVEILLANCE REQUIREMENTS

4.5.3.1 ~~The each ECCS subsystem train shall be demonstrated OPERABLE per the applicable Surveillance Requirements of Specifications 4.5.2., 4.5.2a, 4.5.2b 1), 4.5.2d, 4.5.2f, and 4.5.2g 2).~~

03-10-LS6

4.5.3.2 All centrifugal charging pumps and Safety Injection pumps, except the above allowed OPERABLE pumps, shall be demonstrated ~~inoperable not capable of injection into the RCS~~ at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is ~~less than or equal to 270°F, below the temperature where LTOP is required as specified in the PTLR, by verifying that the motor circuit breakers D.C. control power is de-energized.~~

03-02-LS4

03-11-LG

~~*An inoperable pump may be made OPERABLE for testing or for filling accumulators provided the discharge of the pump has been isolated from the RCS by an isolation valve with power removed from the valve operator, or by a sealed closed manual isolation valve.~~

03-02-LS4

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EMERGENCY CORE COOLING SYSTEMS

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The Refueling Water Storage Tank (RWST) shall be OPERABLE ~~with:~~

- a. A minimum contained borated water volume of 400,000 gallons. 05-03-A
(~~≥81.5% indicated level~~)
- b. A boron concentration of between 2300 and 2500 ppm, and
- c. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

- a. With the RWST inoperable ~~due to boron concentration not within limits or RWST borated water temperature less than the minimum required temperature~~, restore the tank to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. 05-04-LS12
- b. With the RWST inoperable for reasons other than boron concentration ~~not within limits~~, ~~ACTION a~~ restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. 05-04-LS12

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

- a. At least once per 7 days by:
 - 1) Verifying the contained borated water volume in the tank and
 - 2) Verifying the boron of the water.
- b. At least once per 24 hours by verifying the RWST temperature when the outside ambient air temperature is less than 35°F.

Note...RCP seal water requirements are part of current DCPD Technical Specification 3.4.6.2. Operational Leakage. NUREG-1431, Rev. 1 has a specific T.S. 3.5.5 to address RCP seal requirements. ITS 3.5.5 replaces the CTS.

Seal Injection Flow
3.5.5

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LC0 3.5.5 Reactor coolant pump seal injection flow shall be \leq [40] gpm with [centrifugal charging pump discharge header] RCS pressure \geq [2480-2215] psig and \leq [2255] psig and the [charging flow] control valve full open.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Seal injection flow not within limit.	<p>A.1 Verify \approx 100% flow equivalent to a single OPERABLE ECCS charging train is available</p> <p><u>AND</u></p> <p>A.2 Adjust manual seal injection throttle valves to give a flow within limit with [centrifugal charging pump discharge header] RCS pressure \geq [2480-2215] psig and \leq [2255] psig and the [charging flow] control valve full open.</p>	72 hours
B. Required Action and associated Completion Time not met.	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Be in MODE 4.</p>	<p>6 hours</p> <p>12 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
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Seal Injection Flow
3.5.5

<p>SR 3.5.5.1 -----NOTE----- Not required to be performed until 4 hours after the Reactor Coolant System pressure stabilizes at \geq 2215 psig and \leq 2255 psig. ----- Verify manual seal injection throttle valves are adjusted to give a flow within limit with [centrifugal charging pump discharge header] RCS pressure \geq 2400 2215 psig and \leq 2255 psig and the [charging flow] control valve full open.</p>	<p>31 days</p>
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Methodology For Mark-Up of Current TS

This enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (CTS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The CTS are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. **Deletions** - Material is no longer in the specifications. (this includes material which is moved to the BASES of the TS).
2. **Additions** - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the CTS.
3. **Modifications** - This includes requirements which exist in the CTS but are being revised in the improved TS.
4. **Administrative** - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is :

Deletions -

The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.

Additions -

The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.

Modifications -

The information being revised is annotated in the CTS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.

Administrative -

The text of the CTS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the CTS, a change number is provided in the markup of the CTS and an explanation is provided in

Methodology For Mark-up of Current TS
(continued)

Enclosure 3A which explains where that requirement has been located in the improved TS.

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 04-13-LS. The first number (i.e., 04 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS markup. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers	(1 Page)
Description of Changes	(6 Pages)

TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS					
SECTION 3/4.5					
Technical Specification Title	CHG NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Accumulators	01	3.5.1	3.5.1	3.5.1	3.5.1
ECCS Subsystems $T_{avg} > 350^{\circ}\text{F}$	02	3.5.2	3.5.2	3.5.2	3.5.2
ECCS Subsystems $T_{avg} < 350^{\circ}\text{F}$ ECCS Subsystems	03	3.5.3	3.5.3	3.5.3.1	3.5.3
ECCS Subsystems $T_{avg} < 350^{\circ}\text{F}$ Injection Pumps	04	N/A	N/A	3.5.3.2	N/A
ECCS Subsystems $T_{avg} < 200^{\circ}\text{F}$ ECCS Subsystems	04	3.5.4	3.5.4	N/A	N/A
Refueling Water Storage Tank	05	3.5.5	3.5.5	3.5.4	3.5.5

DESCRIPTION OF CHANGES TO CURRENT TS (CTS) SECTION 3/4.5

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	M	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-03	A	Replaces reference to the "pressurizer pressure" with a reference to the "RCS pressure." ACTIONS A. and B. require reducing pressurizer pressure to less than 1000 psig. However, pressurizer pressure instrumentation does not have the range to read that pressure. Consequently, reactor coolant system (RCS) pressure instrumentation is used. For the purposes of this limiting condition of operation (LCO), the use of RCS pressure is equivalent.
01-04	LS 8	The ACTION statement is restructured in agreement with NUREG-1431. The CTS ACTION time (within 1 hour) for restoration of accumulator OPERABILITY for conditions other than a closed isolation valve has been replaced. The replacement ACTION requires that an accumulator inoperable due to boron concentration not within limits must be restored within 72 hours. Also, the requirement to go to HOT SHUTDOWN within the following 6 hours is replaced by the ACTION to reduce RCS pressure to less than or equal to 1000 psig within the following 6 hours. These changes are considered to be relaxations.
01-05	LS 9	The ACTION statement is restructured in agreement with NUREG-1431. The CTS ACTION time (immediately) for restoration of accumulator OPERABILITY due to the isolation valve closed is replaced. The replacement ACTION requires that for reasons other than boron concentration, the accumulator must be restored within 1 hour. Also, the requirement to go to HOT SHUTDOWN within the following 6 hours is replaced by the ACTION to reduce RCS pressure to less than or equal to 1000 psig. These changes are considered to be relaxations.
01-06	A	The words "with three accumulators OPERABLE and" are added to both ACTION statements to make entry into LCO 3.0.3 mandatory with two or more accumulators inoperable. This change is consistent with NUREG-1431 and is considered administrative in nature since it reflects current plant practice, i.e., current ACTION Statements A. and B. are not entered at the same time on different accumulators.

CHANGE NUMBER

NSHC

DESCRIPTION

01-07	LG	The Surveillance Requirement (SR) currently requires a 6 hour surveillance if the makeup source is the refueling water storage tank (RWST) and the RWST has not been diluted since verifying its boron concentration per the RWST LCO. The proposed change would move the statement "and the RWST has not been diluted since verifying ..." from the accumulator SR to the ITS SR 3.5.1.4 Bases. This level of detail is not included in the ISTS and is consistent with the kind of information contained in the Bases.
01-08	A	In accordance with NUREG-1431, the LCO accumulator contained solution volume expressed in cubic feet is replaced by percent (%). Also, the surveillance solution volume increase is revised from 1 percent tank volume (101 gallons) to 5.6 percent of narrow range indicated level, equivalent of 1 percent tank volume. This value can be read from control room indicators. This change in terms is administrative and does not result in a change in the measured volume.
02-01	LG	Consistent with NUREG-1431, the LCO and ACTION a. are revised to replace the word "subsystem" with the word "train." The descriptive information in the LCO is moved to the Bases. Whereas there is no technical change associated with the replacement of the term "subsystem," "train" better describes that all parts of the required system (e.g., piping, instruments, controls, etc.) must be OPERABLE to support the required safety function.
02-02	LS 1	Consistent with NUREG-1431, a Note with respect to RCS pressure isolation valve (PIV) testing is added to the LCO. Plant design requires closure of certain valves in the safety injection (SI) paths to perform PIV testing. Isolation of the injection paths in MODE 3 is currently prohibited as it would constitute entering TS 3.0.3 since both SI trains would be made administratively inoperable. The flow paths are readily restorable from the control room and a single active failure is not likely in the short term (2 hours). The new Note will allow closing these valves for testing without declaring either SI train inoperable. This change is consistent with Industry Traveler TSTF-153.
02-03	LS 2	This change revises ACTION a. to allow for increased flexibility in plant operations under circumstances where components in opposite trains are inoperable, but at least 100 percent of the emergency core cooling system (ECCS) flow equivalent to a single OPERABLE ECCS train is available. Due to the design of the ECCS subsystems, the inoperable condition of one or more components in each train does not necessarily render the ECCS inoperable for performing its safety function. The allowed outage time (AOT) of 72 hours is unchanged; but it is to be contingent on being capable of providing 100 percent of the ECCS flow equivalent to a single OPERABLE ECCS train. This change is consistent with NUREG-1431.

CHANGE NUMBER

NSHC

DESCRIPTION

02-04	TR 2	Consistent with NUREG-1431, the requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted. This change is acceptable because the requirement to submit a report is sufficiently addressed by the reporting requirements contained in 10 CFR 50.73.
02-05	LS 3	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-06		Not used.
02-07	A	Consistent with NUREG-1431, this change revises the surveillance to make it clear that "listed" valve position is the concern and not indicated position in the control room. The surveillance can be satisfied using indicated position in the control room but may also be satisfied using local observation. This is an administrative change since the surveillance acceptance criteria are not changed.
02-08	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-09	LG	The visual inspection surveillance performed when establishing containment integrity is moved to a licensee controlled document, consistent with NUREG-1431.
02-10	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-11	TR 1	Consistent with NUREG-1431, the ECCS pump and valve actuation SR is changed to allow the use of an actual signal, if and when one occurs, to satisfy SRs. The specific signals used to actuate the pumps and valves have been moved to the Bases.
02-12	LG	The ECCS pump performance is revised to be consistent with NUREG-1431. The test method and specific data required to verify pump performance is relocated to the Bases. CTS 4.0.5 no longer exists in the ITS. However, the requirement for an Inservice Testing (IST) Program is moved to Section 5.5.8 of the ITS. The IST Program is referenced directly for the frequency of testing.
02-13	TR 3	The CTS allowance, which permits the ECCS throttle valves to be declared OPERABLE without verifying ECCS throttle valve position for four hours following stroke valve testing or maintenance is deleted from the CTS consistent with NUREG-1431. The ECCS throttle valves are manual valves and plant procedures governing the restoration of equipment after maintenance specify verification of correct throttle position prior to declaring the valves OPERABLE. This requirement is inherent to post-maintenance OPERABILITY requirements and removal from the specifications does not affect TS requirements for testing scope or frequency.

CHANGE NUMBER

NSHC

DESCRIPTION

02-14

A

Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).

02-15

LG

The SR for the flow balance test following ECCS modifications is moved to a licensee controlled document. This requirement is not included in NUREG-1431.

02-16

LG

The specific means by which the ECCS piping is assured to be full of water is moved to the Bases. This level of detail is not included in the ISTS and is consistent with the kind of information contained in the Bases.

02-17

A

This change adds the phrase "that is not locked, sealed, or otherwise secured in position" with regard to which valves require actuation testing. This change is merely a clarification. Valves that are secured in place, are secured in the position required to meet their safety function. The actuation testing ensures that valves can move to the position that meets their safety function. If the valves are secured in the position that meets their safety function, no testing is necessary.

02-18

LG

Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).

02-19

LG

Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).

03-01

LG

Consistent with NUREG-1431, the LCO is revised to replace the word "subsystem" with the word "train" and the descriptive information in the LCO is moved to the Bases. Whereas, there is no technical change associated with the replacement of the term "subsystem," "train" better describes that all parts of the required system (e.g., piping, instruments, controls, etc.) must be OPERABLE to support the required safety functions.

03-02

LS 4

Consistent with NUREG-1431, the LTOP limitation on ECCS pumps and related surveillances are moved to ITS 3.4.12. The prescriptive wording related to pump OPERABILITY is changed to wording specifically addressing the pump's capability to inject into the RCS. This change is less restrictive on the configuration of the centrifugal charging pump (CPP) and SI pumps but is acceptable because it is consistent with the cold overpressure analysis requirements and still precludes flow to the RCS.

03-03

LS 5

Consistent with NUREG-1431, CTS 3.5.3 ACTION a. terminology is revised and the descriptive information moved to the Bases. The ACTION a. Completion Time for COLD SHUTDOWN due to CCP inoperability is increased by 4 hours, from 20 hours to 24 hours. This time is reasonable based on operating experience to reach MODE 5 in an orderly manner, without challenging plant systems or operators, and is consistent with other shutdown ACTION Completion Times to reach MODE 5 from MODE 4.

CHANGE NUMBER

NSHC

DESCRIPTION

03-04

LG

Consistent with NUREG-1431, the ACTION b terminology is revised. The requirement to restore at least one ECCS subsystem is revised to "immediately initiate action to restore" a residual heat removal (RHR) subsystem. With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, when the only available heat removal system is the RHR. Therefore, the appropriate ACTION is to initiate measures to restore one ECCS RHR subsystem and to continue the ACTIONS until the subsystem is restored to OPERABLE status.

Also, the alternate requirement (if RHR cannot be restored) to maintain $T_{avg} < 350^{\circ}F$ by use of alternate heat removal methods is descriptive information and is moved to the Bases. The transition to MODE 3 is already prohibited in this scenario by the ECCS specification for MODES 1, 2, and 3.

03-05

TR 2

Consistent with NUREG-1431, the requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted. This change is acceptable because the requirement to submit a report is sufficiently addressed by the reporting requirements contained in 10 CFR 50.73.

03-06

A

Consistent with TSTF-90, a Note is added to the LCO that clarifies an RHR train's ECCS function is operable if it is capable of being manually realigned to the ECCS mode of operation. This is an administrative change to provide clarification.

03-07

M

Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).

03-08

A

Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).

03-09

M

This change is not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).

03-10

LS 6

Consistent with NUREG-1431, the requirement to demonstrate ECCS train OPERABILITY in MODE 4 in SR [4.5.3.1] has been revised to delete the 31 day surveillance to verify the correct position of each valve in the ECCS flow path which is not already locked in place, and the 18 month surveillance to verify automatic actuation of ECCS pumps and automatic valves.

03-11

LG

The minimum RCS temperature limit below which the CCP and SI pumps must be demonstrated not capable of injecting into the RCS is replaced by the statement "below the temperature where LTOP is required as specified in the pressure temperature limits report (PTLR)." The minimum temperature is a plant specific requirement based on the reactor vessel material characteristics documented in the PTLR and is periodically reviewed and adjusted as required. Referring to the PTLR for the current value is consistent with the relocation of the pressure temperature limits from ITS Section 3.4.3 to

the PTLR.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
03-12	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-01	LS 4	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-04	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-05	M	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-06	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-01	LS 7	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-02	LS 10	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
05-03	A	This change converts the RWST volume requirement from gallons to the equivalent percent of tank water level, in accordance with NUREG-1431. This change in method of indicating tank volume is administrative and does not result in a change in volume of the tank contents.
05-04	LS 12	This change modifies 3.5.5 ACTION a. to include the requirement for RWST borated water temperature to be above the minimum required temperature and ACTION b. to reference ACTION a. The ACTION for water temperature was in ACTION b with a Completion Time of 1 hour. With the water temperature included in ACTION a. the Completion Time is 8 hours. This change is consistent with NUREG-1431. This change from 1 to 8 hours is a relaxation.

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(7 Pages)

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.5

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 M	The percent (%) indicated tank level is added to clarify the volume. Accumulator tank level and tank pressure are revised to provide limits that can be directly confirmed using control room instruments.	No, the CTS values are retained.	Yes.	No, the CTS values are retained.	No, the CTS values are retained.
01-02 A	The nominal tank volume increase is revised to be in the units of indication in the control room.	No, already in the CTS.	Yes	No, already in the CTS.	No, already in the CTS.
01-03 A	The reference to "pressurizer pressure" is replaced by "RCS pressure."	Yes	Yes	Yes	No, already in current TS.
01-04 LS 8	The DCPD ACTION time for restoration of accumulator OPERABILITY for other than a closed isolation valve is increased to 72 hours and the requirement to go to HOT SHUTDOWN is changed to require reduction of RCS pressure to \leq 1000 psig.	Yes	No	No	No
01-05 LS 9	The DCPD ACTION time for restoration of accumulator OPERABILITY for reasons other than boron concentration is increased to 1 hour and the requirement to go to HOT SHUTDOWN is changed to require reduction of RCS pressure to \leq 1000 psig	Yes	No	No	No
01-06 A	This changes adds the words "with three accumulators OPERABLE and" to both ACTION statements to make entry into LCO 3.0.3 mandatory with 2 or more accumulators inoperable.	Yes	Yes	Yes	Yes
01-07 LG	The CTS requires a 6 hour surveillance if the makeup source is the RWST and the RWST has been diluted since verifying its boron concentration per the RWST LCO. This change moves the statement "and the RWST has not been diluted since verifying . . ." from the accumulator SR to the ITS SR 3.5.1.4 Bases.	Yes	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-08 A	The accumulator LCO solution volume for DCP, expressed in cubic feet, is replaced by % tank volume. Also, the surveillance solution volume increase is revised from 1% tank volume (101 gallons) to 5.6% of narrow range indicated level, equivalent of 1% tank volume.	Yes	No	No	No
02-01 LG	The LCO and ACTION a. are revised from subsystem to train and the descriptive information in the LCO moved to the Bases.	Yes	Yes	Yes	Yes
02-02 LS 1	This change allows isolating both SI flow paths for up to 2 hours to perform PIV testing in MODE 3 without declaring either SI train inoperable.	Yes	Yes	Yes	Yes
02-03 LS 2	The change revises ACTION a. to address circumstances where 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available.	Yes	Yes	Yes	Yes
02-04 TR 2	The requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted.	Yes	Yes	Yes	Yes
02-05 LS 3	This change allows operation in MODE 3 pursuant to the LCO for ECCS subsystems $\leq 350^{\circ}\text{F}$ until "all" cold legs exceed the RCS LTOP temperature setpoint in lieu of "one or more."	No, not in the CTS.	Yes	Yes	Yes
02-06	Not used.	N/A	N/A	N/A	N/A
02-07 A	This change clarifies that the surveillance can be satisfied using indicated valve position in the control room but may also be satisfied using local observation.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.5

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-08 A	The accumulator discharge valves and their note are functionally part of the ECCS accumulator subsystem covered by ITS 3.5.1.	No, this note is not in the CTS.	Yes	No, this note is not in the CTS.	No, this note is not in the CTS.
02-09 LG	The visual inspection surveillance performed when establishing containment OPERABILITY is moved to a licensee controlled document.	Yes, moved to The FSAR.	Yes, moved to the TRM.	Yes, moved to Chapter 16 of the USAR.	Yes, moved to FSAR Section 16.5.
02-10 A	The CTS SR for verifying interlock action of the RHR system is moved to ITS SR 3.4.14.2.	No, this SR is not in the CTS.	Yes	Yes	Yes
02-11 TR 1	The ECCS pump and valve actuation SR is changed to allow the use of an actual signal to satisfy SRs.	Yes	Yes	Yes	Yes
02-12 LG	The test method and specific data required to verify ECCS pump performance is moved to the Bases.	Yes	Yes	Yes	Yes
02-13 TR 3	The CTS allowance, which permits the ECCS throttle valves to be declared OPERABLE without verifying ECCS throttle valve stop position for 4 hours following valve stroke testing or maintenance, is deleted from the CTS.	Yes	Yes	Yes	Yes
02-14 A	The note providing a one time SR extension is deleted.	No, not in CTS.	Yes.	No, not in CTS.	Yes
02-15 LG	The SR for the flow balance test following ECCS modifications is moved to a licensee controlled document.	Yes, moved to FSAR.	Yes, moved to TRM.	Yes, moved to USAR Chapter 16.	Yes, moved to FSAR Chapter 16.5.
02-16 LG	The method for ensuring the ECCS system is full of water is moved to the Bases.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.5

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-17 A	This change adds the phrase "that is not locked, sealed, or otherwise secured in position" with regard to which valves require actuation testing.	Yes	Yes	Yes	Yes
02-18 LG	The CPSES requirement for venting the ECCS pump casing and piping following maintenance or activity which drains portions of the system is moved out of the TS.	No, not in CTS.	Yes	No, not in CTS.	No, not in CTS.
02-19 LG	This change moves the requirement that the 18 month verification of automatic ECCS valve actuation and ECCS pump actuation be performed during shutdown to the Bases.	No, DCPD does not have this restriction.	No, CPSES does not have this restriction.	Yes	Yes
03-01 LG	The LCO is revised from subsystem to train and the descriptive information moved to the Bases.	Yes	Yes	Yes	Yes
03-02 LS 4	The LTOP limitation on ECCS pumps and related surveillances are moved to Section 3.4.12 in the ITS. The prescriptive wording related to pump OPERABILITY is changed to wording specifically addressing the pump's capability to inject into the RCS.	Yes	Yes	Yes	Yes
03-03 LS 5	LCO 3.5.3 ACTION a. descriptive information is moved to the Bases. The Completion Time for COLD SHUTDOWN due to CCP inoperability is increased from 20 to 24 hours.	Yes	Yes	Yes	Yes
03-04 LG	The LCO ACTION terminology is revised and the descriptive information moved to the Bases.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.5

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-05 TR 2	The requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted.	Yes	Yes	Yes	Yes
03-06 A	A note is added to the LCO that clarifies an RHR train's ECCS function is OPERABLE if it is capable of being manually realigned to the ECCS mode of operation.	Yes	Yes	Yes	Yes
03-07 M	The surveillance frequency to verify a maximum of one CCP capable of injecting into the RCS is changed from "at least once per 31 days thereafter," to "at least once per 12 hours thereafter."	No, the once per 12 hour surveillance frequency is in the CTS.	No, see CN 03-09-M.	Yes	Yes
03-08 A	A footnote is added to SR [4.5.3.1.1] indicating that the CTS SR to verify the RHR interlock action is not applicable when the RHR suction isolation valves are open to satisfy LCO [3.4.8.3].	No, not in CTS.	Yes	Yes	Yes
03-09 M	The surveillance frequency to verify a maximum of two CCPs capable of injecting into the RCS is changed from within 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever occurs first and once per 31 days, to once per 12 hours.	No, not in CTS.	Yes	No, see CN 03-07-M.	No. see CN 03-07-M.
03-10 LS 6	This change deletes the 31 day surveillance to verify position of valves in the ECCS flow path, and the 18 month surveillance to verify automatic actuation of ECCS pumps and automatic valves. (MODE 4)	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.5

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-11 LG	The minimum RCS temperature limit for DCPD below which the CCPs and the SI pumps must be demonstrated not capable of injecting into the RCS is replaced by the statement "below the temperature where LTOP is required as specified in the PTLR."	Yes	No	No	No
03-12 A	The SR to verify that no more than one CCP and no SI pumps are capable of injecting into the RCS and the SR exception for 4 hours after entering MODE 4 from MODE 3 or until the temperature of one or more RCS cold legs decreases below 325°F, whichever comes first are moved to ITS SR 3.4.12.1, SR 3.4.12.2, and LCO 3.4.12, Note 2.	No, not in CTS .	No, see Change No. 04-06-A.	Yes	Yes
04-01 LS 4	The requirement for having ECCS pump injection sources in excess of that allowed by cold overpressure analysis assumptions be rendered inoperable, has been revised to preclude those pumps from injecting into the RCS.	No, DCPD does not have this TS.	Yes	Yes	Yes
04-02 M	The ACTION required if ECCS pumps in violation of the cold overpressure analyses are capable of injecting into the RCS has been changed to require immediate ACTION initiation. Otherwise, if precluded from compliance, to depressurize the RCS and establish the necessary vent path within 8 hours.	No, DCPD does not have this TS.	Yes	Yes	Yes
04-03 M	This change requires the verification that the disallowed ECCS pumps are not capable of injecting to the RCS on a 12 hour frequency. Previously a 31 day verification on breaker position was required.	No, DCPD does not have this TS.	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.5

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
04-04 A	The Applicability Note regarding SI pump OPERABILITY in MODES 5 and 6 to protect the decay heat removal function has been moved to LCO Note 3 of ITS 3.4.12.	No, DCPD does not have this TS.	No, CPSES current TS does not have this note.	Yes	Yes
04-05 M	The 4 hour AOT for completing actions to make one CCP incapable of injecting, per current licensing basis, has been moved to ITS 3.4.12, LCO Note 1. The four hour AOT for the SI pumps has been deleted.	No, DCPD does not have this TS.	No, CPSES current TS does not have this note.	Yes	Yes
04-06 A	The existing requirement to verify that the SI pump's motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3 is no longer necessary.	No, DCPD does not have this TS.	Yes	No, see change No. 3-12-A.	No, see change No. 3-12-A.
05-01 LS 7	The Completion Time to restore boron concentration or borated water temperature to within limits is increased from 1 to 8 hours.	No, the 8 hour completion time to restore boron concentration is in the CTS.	Yes	Yes	Yes
05-02 LS 10	The shutdown requirement of inoperable RWST would require achieving MODE 3 within "the next" 6 hours.	No, already part of CTS.	Yes	No, already part of CTS.	Yes
05-03 A	This change converts the DCPD RWST volume requirement from gallons to the equivalent percent of tank level with no change in the actual volume.	Yes	No	No	No
05-04 LS 12	This change modifies the ACTION Completion Time to restore RWST water temperature from 1 hour to 8 hours.	Yes	No, already part of CTS.	No, already part of CTS.	No, already part of CTS.

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3..." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION
BASES, FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?
The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:
- a) Increasing the analytical or safety limit,
 - b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
 - c) Increasing the applicability of the specification,
 - d) Providing additional actions,
 - e) Decreasing restoration times,
 - f) Imposing new surveillances, or
 - g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The LCO Applicability is revised to provide a Note with respect to RCS PIV testing in MODE 3. The design requires closure of the PIVs in both SI flow paths to perform PIV testing. Isolation of the injection paths in MODE 3 is currently prohibited as it would constitute entering TS 3.0.3 since both trains of SI would be made administratively inoperable. The revision provides for closing the valves for up to two hours to allow PIV testing. The CCP injection flow paths are available, the SI flow paths are readily restorable from the control room and a single active failure is highly unlikely in the short term. Operating and testing experience has shown that 2 hours is a reasonable period of time required to perform the PIV testing. The low decay heat and non-critical condition of the core allow ample time to take operator action in the event of a loss-of-coolant accident (LOCA) in MODE 3 on startup. The new Note will allow closing these valves without declaring either SI train inoperable. The new Note will allow closing these valves for testing without declaring either ECCS train inoperable.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change does not result in any hardware changes. The ECCS components covered by this TS are not assumed to be initiators of any analyzed event. The proposed change allows both SI flow paths to be isolated for up to 2 hours.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (continued)

The proposed Completion Time of 2 hours is a reasonable time to allow both SI trains to be isolated because of the unlikelihood of a design basis accident (DBA) which would require ECCS during that time period. The ECCS components are not altered by this change, and following the test period, retain their full capability to mitigate the consequences of an accident once correctly aligned. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change will allow both SI trains to be isolated for up to 2 hours. This change does not introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Following the completion of the testing, full SI capability continues. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "LS 1" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The ACTION statement is revised to provide an AOT for 1 or both ECCS trains inoperable. Due to the design of the ECCS subsystems, the inoperable condition of 1 or more components in each train does not necessarily render the ECCS inoperable for performing its safety function. The AOT of 72 hours is unchanged, but it is to be contingent on being capable of providing 100 percent of the ECCS flow equivalent to a single operable ECCS train assuming no single failure. This change is consistent with NUREG-1431.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change does not result in any hardware changes. Providing an AOT for one or both ECCS trains inoperable is not assumed to be an initiator of any analyzed event. The proposed change allows both ECCS trains to be inoperable for up to 72 hours provided the remaining operable ECCS components can provide the flow equivalent to a single operable train which will ensure 100 percent of the flow assumed in the safety analyses, assuming no single failure. The ECCS system function, equivalent of 1 full train, remains available to mitigate the consequences of an unlikely DBA event during this time. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change will allow both ECCS trains to be inoperable for up to 72 hours provided the remaining OPERABLE ECCS components can provide the flow equivalent to a single OPERABLE train. The proposed change does not, introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2 (continued)

safety-related system performs its function. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The allowance to have both ECCS trains inoperable for up to 72 hours, provided the remaining OPERABLE ECCS components can provide the flow equivalent to a single OPERABLE train, does not have any effect on accident or transient analysis. The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "LS2" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The Completion Time to achieve COLD SHUTDOWN due to failure to restore the OPERABILITY of an ECCS CCP subsystem within 1 hour is extended from 20 to 24 hours. This is consistent with NUREG-1431 and is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Due to the low energy, stable conditions associated with operation in MODE 4, the occurrence of a DBA is unlikely. In recognition, the ECCS operational requirements are reduced, with only 1 train of the ECCS CCP subsystem required to be OPERABLE. The CTS has an ACTION Completion Time of 1 hour to restore required ECCS CCP OPERABILITY. The 1 hour restoration Completion Time is in accordance with NUREG-1431, and is not affected by this proposed change, and does not alter the probability of an accident during the ACTION Completion Time. The allowed time to reach MODE 5 is increased from 20 hours to 24 hours by this proposed change. This additional time allows for an orderly MODE transition. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any safety-

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5 (continued)

related system performs its function. Therefore, this proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "LS5" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6

10 CFR 50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, the requirement to demonstrate ECCS train OPERABILITY in MODE 4 has been revised to delete the 31 day surveillance to verify the correct position of each valve in the ECCS flow path which is not already locked in place, and the 18 month surveillance to verify automatic actuation of ECCS pumps and automatic valves. This change is acceptable because the ECCS operational requirements are reduced due to the stable reactivity conditions and limited core cooling requirements associated with operation in MODE 4 and the unlikelihood of occurrence of a DBA in MODE 4. It is understood these surveillance reductions increase the possibility that automatic positioning of ECCS valves and actuation of ECCS pumps may not occur on a SI actuation signal. In MODE 4, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Due to the stable conditions associated with operation in MODE 4, the probability of occurrence of a DBA is lower than in MODES 1, 2, and 3. Because of the reduced core cooling requirements, sufficient time is available for manual actuation/alignment of required ECCS equipment. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change will delete certain surveillances for MODE 4 OPERABILITY of an ECCS train. However, the proposed change does not introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "LS6" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The ACTION statement is revised in accordance with NUREG-1431 to extend the time required to restore accumulator water boron concentration to within limits from 1 to 72 hours. A 1 hour ACTION to initiate plant shutdown compromises the opportunity for verification, diagnosis, and restoration of the condition of the accumulator to within limits. Changes in boron concentration are slow and the ACTION change provides a more reasonable time in which to restore limits. Increasing the AOT from 1 hour to 72 hours could avoid unnecessary plant transients and plant shutdowns if OPERABILITY cannot be restored within 1 hour but could be restored within 72 hours, thus improving plant safety and increasing plant availability.

The boron in the accumulator water contributes to the analyses assumption that the combined ECCS water in the partially recovered core during the early reflood phase of a large break LOCA is sufficient to keep that portion of the core subcritical. Core cooling during the ECCS injection mode is not affected by the accumulator water boron concentration being outside limits because the accumulator water remains available for injection throughout the duration of the AOT.

The condition of 1 accumulator not within the boron concentration limits does not affect the other 3 accumulators as the accumulators are mutually isolated. One accumulator below the minimum boron concentration will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Consequently, although the accumulator may be inoperable in accordance with TS requirements due to boron concentration outside of limits, it still retains capability to contribute in satisfying its safety function. If 2 or more accumulators are inoperable for any reason, ITS 3.5.1, ACTION D requires immediate entry into TS 3.0.3.

Increasing the AOT is consistent with recommendations of NUREG-1024, "Technical Specification -- Enhancing the Safety Impact." NUREG 1024 states:

"Allowable outage times that are too short will subject the plant to unnecessary trips, transients and fatigue cycling. Outage times that are too short also may result in less thorough repair and post-repair testing before equipment is returned to service."

Replacing the ACTION requirement to go to HOT SHUTDOWN within the following 6 hours with the requirement to reduce RCS pressure to less than or equal to 1000 psig within the following 6 hours makes the ACTION consistent with the Applicability. Reducing the RCS pressure to less than or equal to 1000 psig results in the accumulators no longer being required, and TS Applicability can be exited. At pressures less than or equal to 1000 psig, the rate of RCS blowdown is reduced and the ECCS pumps can provide adequate injection to ensure the peak clad temperature remains below the acceptance limit of 10 CFR 50.46.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8
(continued)

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change does not result in any hardware or operating procedure changes. The ECCS conditions covered by this TS are not assumed to be initiators of any analyzed event. The potential for the accumulator water boron concentration to be outside limits is small because the accumulators and contents are not involved with normal plant operation and are not subject to process variations associated with plant operation. The proposed change allows a longer time to restore conditions to within limits. The reactor will be in MODES 1, 2, 3 (with RCS pressure >1000 psig) during this condition.

The effect of a departure from boron concentration is insignificant to core criticality, whereas the effect of the water to mitigate consequences is not decreased. Consequently, although the likelihood of an event occurrence during the additional 71 hours is small, the consequences of an event occurring during the period are not significantly different.

The replacement of the requirement to go to HOT SHUTDOWN with the requirement to go to a RCS pressure less than or equal to 1000 psig reduces the plant transients. Reducing the RCS pressure to less than or equal to 1000 psig results in the accumulators no longer being required for TS Applicability. However the peak clad temperature remains below the acceptance limit of 10 CFR 50.46 using ECCS pump injection without the contribution from the accumulators. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change will allow additional time to restore boron concentration to within limits. The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed), does not change the method by which any safety-related system performs its function, and does not change parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The additional time to restore boron concentration to within limits prior to requiring a unit shutdown is based on the fact that the contents of the affected accumulator are still available for injection in agreement with the analyses assumptions. Consequently, any change in a margin of safety will be insignificant and offset by the benefit of avoiding an unnecessary plant transient.

The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "LS8" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The ACTION statement is revised in agreement with NUREG-1431 to extend the time required to restore accumulator OPERABILITY for reasons other than boron concentration not within limits from immediately to 1 hour. Accumulator OPERABILITY is important to satisfy the analyses assumption that the contents of 3 of the 4 accumulators reach the core following a LOCA. However, an immediate ACTION to initiate plant shutdown denies the opportunity for verification, diagnosis, and restoration of the condition of the accumulator to within limits. The additional time proposed will reduce the probability of unnecessary plant transients and plant shutdowns, thus improving plant safety and increasing plant availability. If 2 or more accumulators are inoperable for any reason, ITS 3.5.1 ACTION D requires immediate entry into TS 3.0.3.

Increasing the AOT is consistent with recommendations of NUREG-1024, "Technical Specification -- Enhancing the Safety Impact." NUREG 1024 states:

"Allowable outage times that are too short will subject the plant to unnecessary trips, transients and fatigue cycling. Outage times that are too short also may result in less thorough repair and post-repair testing before equipment is returned to service."

Replacing the ACTION requirement to go to HOT SHUTDOWN within the following 6 hours with the requirement to reduce RCS pressure to less than or equal to 1000 psig within the following 6 hours makes the ACTION consistent with the Applicability. Reducing the RCS pressure to less than or equal to 1000 psig results in the accumulators no longer being required for TS Applicability. At pressures less than or equal to 1000 psig, the rate of RCS blowdown is reduced and the ECCS pumps can provide adequate injection to ensure the peak clad temperature remains below the acceptance limit of 10 CFR 50.46.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9 (continued)

This change does not result in any hardware or operating procedure changes. The ECCS conditions covered by this TS are not assumed to be initiators of any analyzed event. The 1 hour Completion Time ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The likelihood of an event occurrence during 1 hour is low.

The replacement of the requirement to go to HOT SHUTDOWN with the requirement to go to a RCS pressure less than or equal to 1000 psig reduces plant transients. Reducing the RCS pressure to less than or equal to 1000 psig results in the accumulators no longer being required for TS Applicability. However, the peak clad temperature remains below the acceptance limit of 10 CFR 50.46 using ECCS pump injection without the contribution from the accumulators. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change will allow 1 hour to restore an accumulator to OPERABLE status. The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed), does not change the method by which any safety-related system performs its function, and does not change parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The occurrence of an event during the 1 hour AOT is extremely unlikely, whereas allowing some time to restore the accumulator may avoid unnecessary plant transients associated with a required shutdown. As such, any change in a margin of safety will be insignificant and offset by the benefit of reducing the possibility of an unnecessary plant transient. The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "LS9" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS12

10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The ACTION statement is revised to extend the time required to restore RWST borated water temperature to within limits from 1 to 8 hours. Changes in water temperature are slow and the change provides a more reasonable time in which to restore limits. The increase in allowed time to restore conditions to within limits prior to requiring a unit shutdown recognizes that the deviation from the analysis temperature assumption would probably be minor due to the moderate site temperature and that the contents of the tank are still available for injection. Also, the occurrence of an event requiring injection of the RWST contents during the increased Completion Time is extremely unlikely. The additional time proposed will reduce the possibility of unnecessary plant transients and plant shutdowns, thus improving plant safety and increasing plant availability.

Increasing the AOT is consistent with recommendations of NUREG-1024, "Technical Specification - Enhancing the Safety Impact." NUREG 1024 states:

"Allowable outage times that are too short will subject the plant to unnecessary trips, transients and fatigue cycling. Outages times that are too short also may result in less thorough repair and post-repair testing before equipment is returned to service."

The contents of the tank are still available for injection during the Completion Time and the extent of temperature outside the limit is not likely to be significant to safety. This change is consistent with NUREG-1431.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS12 (continued)

This change does not result in any hardware or operating procedure changes. The RWST conditions covered by this TS are not assumed to be initiators of any analyzed event. The likelihood of an event occurrence during the additional 7 hours is very small, and the analysis effect of small departures from temperature is judged to be slight. The consequences of an event occurring during the increased Completion Time period are not significantly different. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change will allow additional time to restore conditions to within limits; but it does not alter the activity. The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed), does not change the method by which any safety-related system performs its function, and does not change parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not significantly changed. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "LS12" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR1

10 CFR 50.92 EVALUATION FOR RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision modifies the CTS to additionally allow the use of actual actuation signals for surveillances that currently call for testing using simulated test signals only. This change achieves consistency with the STS (NUREG-1431).

In several specifications throughout the CTS, OPERABILITY of certain equipment is demonstrated by ensuring that the equipment performs its safety function upon receipt of a simulated test signal. The intent of a 'simulated' signal was to be able to perform the required testing without the occurrence (or without causing) an actual signal generating event. However, the unintended effect was to require the performance of the surveillance (using a test signal) even if an actual signal had previously verified the operation of the equipment. This change allows credit to be taken for actual events when the required equipment actuates successfully.

While the occurrence of events that cause actuation of accident mitigation equipment is undesirable, the actuation of mitigation equipment on an actual signal is a better demonstration of its OPERABILITY than an actuation using a test signal. Thus, the change does not reduce the reliability of the equipment tested. The change also improves plant safety by reducing the amount of time the equipment is taken out of service for testing, and thereby increasing its availability during an actual event, and by reducing the wear of the equipment caused by unnecessary testing.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change allows the use of an actual actuation signal (when/if it occurs) to satisfy SRs currently requiring simulated test signals to demonstrate equipment OPERABILITY. While the change takes advantage of events that may have occurred, it has no adverse effect on any accident initiators or accident consequences.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR1 (continued)

It may also reduce accident consequences by increasing the equipment availability (i.e., less time in test). Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The use of an actual actuation signal to satisfy a SR has either no impact on, or increases the margin of plant safety by:

- a) Increasing mitigation equipment availability, and
- b) Improving mitigation equipment reliability by potentially reducing wear caused by unnecessary testing.

The proposed change does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "TR1" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR2

10 CFR 50.92 EVALUATION FOR RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change in accordance with NUREG-1431, removes the requirement for a special report to be prepared and submitted to the NRC within 90 days of an ECCS actuation and injection event. Reporting to the NRC will be done commensurate with the reporting requirements of 10 CFR 50.72 and 50.73.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This proposed change is an administrative reporting change and cannot affect any accident probability or consequences or the method by which the plant is operated. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change is administrative and does not introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change is administrative and does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

**NSHC TR2
(continued)**

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "TR 2" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR3

10 CFR 50.92 EVALUATION FOR RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed change removes references to specific post-maintenance tests from the CTS. Post-maintenance testing programs are controlled via plant administrative procedures in accordance with licensee controlled document (ITS Section 5.4.1, Final Safety Analysis Report and Operating Quality Assurance Program) commitments to NRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," and ANS 3.2-ANSI N18.7, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." Specific post-maintenance testing requirements are contingent on the type and scope of maintenance actually performed as well as the availability and viability of test equipment, techniques, etc. Removal of specific testing requirements from the CTS and reliance on normal post-maintenance testing programs addressed by licensee controlled documents allow flexibility to modify testing to address the circumstances of the maintenance performed while still assuring OPERABILITY of equipment returned to service.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This proposed change is an administrative change which removes specific post-maintenance test requirements from the CTS. The testing, or equivalent testing, to assure equipment OPERABILITY prior to return to service would still be done as required by normal plant maintenance retest programs. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR3 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change is administrative and does not introduce new equipment, does not involve any physical alteration to any plant equipment, and does not involve any changes in the method by which any safety-related system performs its function. Therefore, this proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change is administrative and does not alter the basic regulatory requirements and does not change any assumptions, conditions, or acceptance criteria of any analyzed event. The analyses remain valid and the margin of safety is not changed. Therefore, this proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, the activities associated with NSHC "TR 3" resulting from the conversion to the ITS format are seen to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers (1 Page)

NUREG-1431 Specifications that are not applicable (1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
3.5.1.	3.5-1
3.5.2.	3.5-4
3.5.3.	3.5-8
3.5.4.	3.5-10
3.5.5.	3.5-12

Methodology (2 Pages)

Industry Travelers Applicable to Section 3.5

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-90, Rev. 1	Incorporated	3.5-6	
TSTF-117	Incorporated	3.5-1	
TSTF-153	Incorporated	3.5-8	
TSTF-155	Not Incorporated	N/A	Not NRC approved as of traveler cut-off date.
WOG-84	Incorporated	3.5-4	DCPP and CPSES only.

NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

<u>Specification #</u>	<u>Specification Title</u>	<u>Comments</u>
3.5.6	Boron Injection Tank (BIT)	Not part of plant design

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 ~~{Four}~~ ECCS accumulators shall be OPERABLE.

B

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer RCS pressure > ~~{1000}~~ psig.

3.5-1

B

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Reduce pressurizer RCS pressure to {1000} psig.	12 hours <u>3.5-1</u> <u>B</u>
D. Two or more accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.1.1 Verify each accumulator isolation valve is fully open.	12 hours

SURVEILLANCE		FREQUENCY	
SR 3.5.1.2	Verify borated water volume in each accumulator is \geq [7853 gallons] 60% and \leq [8171 gallons] 72.6%	12 hours	<u>B-PS</u>
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is \geq [385 595.5] psig and \leq [481 647.5] psig	12 hours	<u>B-PS</u>
SR 3.5.1.4	Verify boron concentration in each accumulator is \geq [1900 2200] ppm and \leq [2100 2500] ppm.	31 days <u>AND</u> -----NOTE----- Only required to be performed for affected accumulators ----- Once within 6 hours after each solution volume increase of \geq [gallons] 5% of narrow range indicated level that is not the result of addition from the refueling water storage tank	<u>B-PS</u> <u>PS</u>
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when pressurizer RCS pressure is \geq [2000] >1000 psig.	31 days	<u>3.5-1</u> <u>B-PS</u> <u>ED</u>

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

-----NOTES-----

- | | | |
|----|--|------|
| 1. | In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valve(s) for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1. | B-PS |
| 2. | Operation in MODE 3 with ECCS pumps declared inoperable, pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is allowed for up to 4 hours or until the temperature of all RCS cold legs exceeds [375] °F, whichever comes first. | B-PS |

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. <u>AND</u> At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed		12 hours
	<u>Number</u>	<u>Position</u> <u>Function</u>	
	8703	Closed RHR to RCS Hot Legs	B-PS
	8802A	Closed Safety Injection to RCS Hot Legs	
	8802B	Closed Safety Injection to RCS Hot Legs	B-PS
	8809A	Open RHR to RCS Cold Legs	
	8809B	Open RHR to RCS Cold Legs	
	8835	Open Safety Injection to RCS Cold Legs	
	8974A	Open Safety Injection Pump Recir. to RWST	
	8974B	Open Safety Injection Pump Recir. to RWST	
	8976	Open RWST to Safety Injection Pumps	
	8980	Open RWST to RHR Pumps	
	8982A	Closed Containment Sump to RHR	
	8982B	Closed Containment Sump to RHR	
	8992	Open Spray Additive Tank to Eductor	
	8701	Closed RHR Suction	
	8702	Closed RHR Suction	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY										
SR 3.5.2.2 Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days										
SR 3.5.2.3 Verify ECCS piping is full of water	31 days <u>B</u>										
SR 3.5.2.4 Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program										
SR 3.5.2.5 Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months <u>B</u>										
SR 3.5.2.6 Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months <u>B</u>										
SR 3.5.2.7 Verify, for each ECCS throttle valve listed below, each mechanical position stop is in the correct position. <table border="1" data-bbox="313 1281 974 1512" style="margin-left: 40px;"> <thead> <tr> <th>Charging Injection Throttle Valves</th> <th>Safety Injection Throttle Valves</th> </tr> </thead> <tbody> <tr> <td>8810A</td> <td>8822A</td> </tr> <tr> <td>8810B</td> <td>8822B</td> </tr> <tr> <td>8810C</td> <td>8822C</td> </tr> <tr> <td>8810D</td> <td>8822D</td> </tr> </tbody> </table>	Charging Injection Throttle Valves	Safety Injection Throttle Valves	8810A	8822A	8810B	8822B	8810C	8822C	8810D	8822D	18 months <u>3.5-3</u> <u>B-PS</u>
Charging Injection Throttle Valves	Safety Injection Throttle Valves										
8810A	8822A										
8810B	8822B										
8810C	8822C										
8810D	8822D										
SR 3.5.2.8 Verify, by visual inspection, each ECCS train containment recirculation sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	18 months <u>B</u> <u>PS</u>										

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS--Shutdown

LCO 3.5.3 One ECCS train shall be OPERABLE.

NOTE
An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation.

3.5-6

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Required ECCS residual heat removal (RHR) subsystem inoperable.</p>	<p>A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status</p>	<p>Immediately</p>
<p>B. Required ECCS high head Centrifugal Charging Pump subsystem inoperable.</p>	<p>B.1 Restore required ECCS high head Centrifugal Charging Pump subsystem to OPERABLE status.</p>	<p>1 hour</p>
<p>C. Required Action and associated Completion Time of Condition B not met.</p>	<p>C.1 Be in MODE 5.</p>	<p>24 hours</p>

B

B-PS

B

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.5.3.1</p> <p style="text-align: center;">----- NOTE -----</p> <p>An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation.</p> <p style="text-align: center;">-----</p> <p>The following SRs are applicable for all equipment required to be OPERABLE:</p> <p>[SR 3.5.2.1] [SR 3.5.2.7] [SR 3.5.2.3] SR 3.5.2.8 SR 3.5.2.4</p>	<p style="text-align: right;"><u>3.5-6</u></p> <p>In accordance with applicable SRs</p> <p style="text-align: right;"><u>B</u> <u>B:PS</u></p>

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. RWST boron concentration not within limits.</p> <p><u>OR</u></p> <p>RWST borated water temperature not within limits.</p>	<p>A.1 Restore RWST to OPERABLE status.</p>	<p>8 hours</p>
<p>B. RWST inoperable for reasons other than Condition A.</p>	<p>B.1 Restore RWST to OPERABLE status.</p>	<p>1 hour</p>
<p>C. Required Action and associated Completion Time not met.</p>	<p>C.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>C.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.5.4.1</p> <div style="border: 1px solid black; padding: 5px; margin: 5px 0;"> <p style="text-align: center;">NOTE</p> <p>Only required to be performed when ambient air temperature is \leq [35] °F or \geq [100] °F.</p> </div> <p>Verify RWST borated water temperature is \geq [35] °F and \leq [100] °F.</p>	<p style="text-align: right;"><u>B-PS</u></p> <p>24 hours</p> <p style="text-align: right;"><u>3.5-8</u></p>
<p>SR 3.5.4.2</p> <p>Verify RWST borated water volume is \geq [466,200 400,000 gallons (81.5% indicated level)]</p>	<p>7 days</p> <p style="text-align: right;"><u>B-PS</u></p>
<p>SR 3.5.4.3</p> <p>Verify RWST boron concentration is \geq [2000 2300] ppm and \leq [2200 2500] ppm.</p>	<p>7 days</p> <p style="text-align: right;"><u>B-PS</u></p>

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LC0 3.5.5 Reactor coolant pump seal injection flow shall be \leq ~~[40]~~ gpm with ~~[centrifugal charging pump discharge header]~~ RCS pressure \geq ~~[2480- 2215]~~ psig and \leq ~~[2255]~~ psig and the ~~[charging flow]~~ control valve full open..

B
3.5.5

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Seal injection flow not within limit.	A.1 Verify \geq 100% flow equivalent to a single OPERABLE ECCS charging train is available	4 hours <u>3.5.4</u>
	A.2 Adjust manual seal injection throttle valves to give a flow within limit with [centrifugal charging pump discharge header] RCS pressure \geq [2480- 2215] psig and [2255] psig and the [charging flow] control valve full open.	72 hours <u>3.5.5</u> <u>B</u>
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.5.5.1 -----NOTE----- Not required to be performed until 4 hours after the Reactor Coolant System pressure stabilizes at \geq [2215 psig and \leq 2255 psig.] ----- Verify manual seal injection throttle valves are adjusted to give a flow within limit with [centrifugal charging pump discharge header] RCS pressure \geq [2480 2215 psig and \leq 2255] psig and the [charging flow] control valve full open.</p>	<p style="text-align: center;">31 days</p> <p style="text-align: right;"> <u>B</u> <u>B</u> <u>3.5-5</u> </p>

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exists in NUREG-1431, Rev. 1, but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is :

Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.

Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.

Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.

Editorial Changes- Changes/corrections which are obviously editorial are annotated using the redline/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.

Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "redline" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line", "strike-out" and margin codes are as follows:

1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is

Methodology For Mark-up of NUREG-1431 Specifications
(continued)

used as the margin code.

2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out", the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.
3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is redlined. The text included within the brackets is not redlined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are struck-out, redlined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note, NUREG-1431, Rev. 1 is used for all markups. Industry Travelers which are incorporated are indicated using the "redlines", "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES

Mark-up:

<u>BASES</u>	<u>PAGE</u>
3.5.1.....	B 3.5-1
3.5.2.....	B 3.5-9
3.5.3.....	B 3.5-20
3.5.4.....	B 3.5-24
3.5.5.....	B 3.5-35
 Methodology	 (1 Page)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply borated water to replace inventory in the reactor vessel during the blowdown refill phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and ejection spill out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by two check valves in series and by an open motor operated isolation valve (8808A, B, C, and D). The motor operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to ensure that the valves will automatically open as RCS pressure increases to above permissive circuit P-11 setpoint. to receive an "open" signal when permissive circuit P-11 is cleared. However, before permissive circuit P-11 is reached, these valves are manually opened and their motor operator breakers are sealed open to satisfy SR 3.5.1.5. Therefore, in the event of a LOCA, accumulator actuation is passive. (Ref [6.1])

BASES

BACKGROUND
(continued)

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. However, if these valves were closed, they would be automatically opened as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE
SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2 and 4). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow with no credit taken for ECCS pump flow until an effective delay has elapsed. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break. No operator action is assumed during the blowdown stage of a large break LOCA.

BASES

APPLICABLE SAFETY
ANALYSES (continued)

~~As a conservative estimate, no credit is taken for ECCS pump flow until an effective delay has elapsed. This delay accounts for the diesels starting and the pumps being loaded and delivering full flow. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. During this time, the accumulators are analyzed as providing the sole source of emergency core cooling. No operator action is assumed during the blowdown stage of a large break LOCA.~~

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the SI pumps begin RCS injection, however, the increase in fuel clad temperature is terminated solely primarily by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and the ECCS centrifugal charging and SI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease. The accumulators do not discharge above the pressure of their nitrogen cover gas (595.5 to 647.5 psig). At higher pressures the ECCS centrifugal charging pumps and SI pumps injection becomes solely responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 3) that are applicable for the accumulators will be met following a LOCA:

- a. Maximum fuel element cladding temperature is $\leq 2200^{\circ}\text{F}$;
- b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium-water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding cladding surrounding the plenum volume, were to react; and
- d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase

BASES

APPLICABLE SAFETY
ANALYSES (continued)

in water volume is a peak clad temperature penalty. For large breaks depending on the NRC-approved methodology used to analyze large breaks, an increase in water volume can result in either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of \geq [6468] 60.8% (836 cubic feet) gallons and \leq [6879] 72.6% (864 cubic feet) gallons as read on narrow range level instruments not including instrument uncertainty. To allow for instrument inaccuracy values of [6520] gallons and [6820] gallons are specified.

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction in below the accumulator LCO minimum boron concentration would produce a subsequent reduction in the available containment recirculation sump boron concentration for post LOCA shutdown and an increase in the maximum sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure (603 psia) (595.5 psig), since sensitivity analyses have demonstrated that higher nitrogen cover pressure results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit (693 psia) (647.5 psig) prevents accumulator relief valve actuation and ultimately preserves accumulator integrity, provides margin to assure inadvertent relief valve actuation does not occur.

These analysis-assumed pressures are specified in the SRs. Volumes are shown on the control board indicators as % readings on accumulator narrow range level instruments. Adjustments to the analysis parameters for instrument inaccuracies or other reasons are applied to determine the acceptance criteria used in the plant surveillance procedures. These adjustments assure the assumed analyses parameters are maintained.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

The accumulators satisfy Criterion 3 of the NRC Policy Statement. 10 CFR 50.36(c)(2)(ii)

BASES

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above [2000] a nominal RCS pressure of 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at RCS pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are normally closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONS

A.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, the ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in

BASES

the core. In addition, current analysis techniques demonstrate that the accumulators will do not discharge following a large main steam line break for the majority of plants. Even if they do discharge, their impact of their discharge is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and pressurizer RCS pressure reduced to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE
REQUIREMENTSSR 3.5.1.1

Each accumulator motor operated isolation valve (8808A, B, C, and D) should be verified to be fully open every 12 hours. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident

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SURVEILLANCE
REQUIREMENTS
(continued)

analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or in-leakage. Sampling the affected accumulator within 6 hours after a solution volume increase of 5.6-1% (101 gallon) narrow range indicated level will identify whether in-leakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), and the RWST has not been diluted since verifying that its boron concentration satisfies SR 3.5.4.3, because the water contained in the RWST is within the accumulator boron concentration requirements as verified by SR 3.5.4.3. This is consistent with the recommendation of GL 93-05 (Ref. 5).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator (8808A, B, C, and D) when the pressurizer RCS pressure is ≥ 2000 greater than 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when pressurizer RCS pressure is < 2000

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~~less than or equal to 1000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. The valves to be closed to enable plant shutdown without discharging the accumulators into the RCS. Even with power supplied to the valves, inadvertent closure is~~

SURVEILLANCE
REQUIREMENTS (continued)

prevented by the RCS pressure interlock (P-11) associated with the valves.

Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. IEEE Standard 279-1971.
 2. FSAR, Chapter ~~6~~.
 3. 10 CFR 50.46.
 4. FSAR, Chapter ~~15~~.
 5. ~~GL 93-05, Item 7.1~~
 6. ~~DCM S 38A~~
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

- a. Loss of coolant accident (LOCA), non-isolable coolant leakage greater than the capability of the normal charging system;
- b. Rod ejection accident;
- c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and
- d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head) and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the Refueling Water Storage Tank (RWST) are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS components are divided into two trains, A and B. The following are the train assignments for the ECCS pumps:

Train A:	RHR Pump 2	Train B:	RHR Pump 1
	SI Pump 1		SI Pump 2
	Centrifugal Charging Pump (CCP) 1		Centrifugal Charging Pump (CCP) 2

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the centrifugal charging pumps, CCPs, the RHR pumps, heat exchangers, and the SI pumps. Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the

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(continued)

ability to utilize components from opposite trains to achieve the required 100% flow to the core.

There are three phases of ECCS operation following a LOCA: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the RWST and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment recirculation sump has enough water to supply the required net positive suction head to the ECCS RHR pumps, suction is switched to the containment recirculation sump for cold leg recirculation. After several hours, the ECCS flow operation is shifted to the hot leg recirculation phase to provide a backflush, which would reduce the boiling in the top of the core and any resulting boron precipitation, reverse flow through the core to backflush out the high boron concentration that could result from core boiling after a cold leg break.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. The RWST header supplies separate piping supplies for each subsystem, and each train within the subsystem. The discharge from the CCPS centrifugal charging pumps combines in a common header and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. Control Throttle/runout valves are set to balance the flow to the RCS. The throttle/runout valves also protect the SI and CCPS from exceeding their runout flow limits. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the CCPS centrifugal charging pumps supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment recirculation sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation discharge is through the same paths as the injection phase to the cold legs. Subsequently, recirculation alternates provides injection between both the hot and cold legs.

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased heat removal events, such as a main steam line break (MSLB). The limiting design conditions

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BACKGROUND
(continued)

occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle.

(continued)

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start after a one second sequencer delay in the programmed time sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

Each ECCS pump is provided with normally open miniflow lines for pump protection. The RHR miniflow isolation valves close on flow to the RCS and have a time delay to prevent them from closing until the RHR pumps are up to speed and capable of delivering fluid to the RCS. The SI pump miniflow isolation valves are closed manually from the control room prior to transfer from injection to recirculation. The CCP miniflow isolation valves are also closed manually from the control room prior to transfer from injection to recirculation.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE
SAFETY ANALYSIS

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

- a. Maximum fuel element cladding temperature is $\leq 2200^{\circ}\text{F}$;
- b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;
- d. Core is maintained in a coolable geometry; and

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APPLICABLE SAFETY
SAFETY ANALYSES
(continued)

- e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post-trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement to limit runoff flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in the injection phase for mitigation of a small break LOCA event. This event establishes the flow and discharge head for the design point of the CCPs centrifugal charging pumps. The SGTR and MSLB events also credit the CCPs centrifugal charging pumps. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

- a. A large break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (all EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and
- b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large break LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small break LOCA to maintain core subcriticality. For smaller break LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of the NRC Policy Statement. 10 CFR 50.36(c)(2)(iii)

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LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem.

Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal, and initiating semi-automatic switchover of suction having its. During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold legs. The ECCS suction is manually transferred to the containment recirculation sump to place the system in the recirculation mode of operation to supply its flow to the RCS hot and cold legs. During the recirculation operation, the RHR pumps provide suction to the charging and SI pumps.

~~During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment recirculation sump and to supply its flow to the RCS hot and cold legs.~~

During recirculation operation, the flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

~~As indicated in Note 1, the SI flow paths may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room and a single active failure (Ref. 7) is not assumed coincident with this testing. Therefore the ECCS trains are considered OPERABLE during this isolation.~~

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

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APPLICABILITY
(continued)

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS - Shutdown."

~~As indicated in Note 1, the flow path may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room. As indicated in Note 2, operation in MODE 3 with ECCS trains declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered inoperable at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.~~

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available (capable of injection into the RCS if actuated) the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design safety function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to

BASES

a single OPERABLE ECCS train remains available. (i.e. minimum of one OPERABLE CCP, SI, and RHR pump and applicable flow paths capable of drawing from the RWST and injecting into the RCS cold legs). This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

The intent of this Condition to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available applies to both the injection mode and the recirculation mode.

ACTIONS
(continued)

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR crossover cross tie valve can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

Opening the containment recirculation sump access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered. (Ref. 9)

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Valve position is the concern and not indicated position in the control room. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power by a control board switch in the correct position ensures that they cannot change position as a result of an active failure or be

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SURVEILLANCE
REQUIREMENTS
(continued)

inadvertently misaligned. The surveillance can be satisfied using indicated position in the control room but may also be satisfied using local observation. These valves are of the type, described in References 6 and 7, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely. As noted in LCO Note 1, both SI pump flow paths may each be isolated for two hours in MODE 3 by closure of one or more of these valves to perform pressure isolation valve testing.

In addition to the valves listed in SR 3.5.2.1, there are other ECCS related valves that must be appropriately positioned. Improper valve position can affect the ECCS performance required to meet the analysis assumptions. These valves are identified in plant documents and are listed in the following table.

ECCS Valve Position Table

Valve Number	Valve Function	Required Valve Position	MODES
8105	CCP 1 and 2 Recirc Line Isolation	Open	1, 2, 3
8106	CCP 1 and 2 Recirc Line Isolation	Open	1, 2, 3
8716A	RHR Cross-tie Line	Open	1, 2, 3
8716B	RHR Cross-tie Line	Open	1, 2, 3
9003A	RHR to Containment Spray	Closed	1, 2, 3
9003B	RHR to Containment Spray	Closed	1, 2, 3
8804A	RHR to CCP	Closed	1, 2, 3
8804B	RHR to SI Pump	Closed	1, 2, 3
8741	RHR to RWST - Manual Valve	Closed	1, 2, 3
SI-1	RWST to ECCS - Manual Valve	Open	1, 2, 3, 4
8923A*	Train "A" SI Pump Suction Valve	Open	1, 2, 3

* Valve can be closed, but not when RHR Train "A" (containing RHR pump 2) is out of service. Closing this valve with RHR Train "A" out of service would result in both trains of ECCS being inoperable due to the ECCS piping configuration.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist

SURVEILLANCE

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REQUIREMENTS
(continued)

for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

With the exception of the operating ~~CCP centrifugal charging pump~~, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of non-condensable gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

~~The intent of the SR is to assure the ECCS piping is full of water. Different means of verification, as alternates to venting the accessible system high points, can be employed to provide this assurance.~~

~~Venting of the accessible ECCS high points prior to entering MODE 3 ensures the system is full of water and will perform properly, injecting its full capacity into the RCS on demand.~~

~~The CCP design and attached piping configuration allow the CCP to vent the accumulated gases via the attached suction and discharge piping. Continuous venting of the suction piping to the Volume Control tank (VCT) and manual venting of the discharge piping high points satisfies the pump casing venting requirements for the CCPs.~~

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by

SURVEILLANCE

BASES

REQUIREMENTS
(continued)

~~Section XI of the ASME Code. (Ref. B) This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to within the performance assumed in the plant safety analysis. SRS are specified in Technical Requirements Manual and in the applicable portions of the Inservice Testing Program, which encompasses Section XI Part 6 of the ASME Code for Operation and Maintenance of Nuclear Power Plants. (Ref. 8). Section XI this section of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.~~

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

~~The correct Realignment position of throttle/runout valves in the ECCS flow paths on an SI signal is necessary for proper ECCS performance. These manual throttle/runout valves are positioned during flow balancing and have mechanical locks and seals -steps to allow ensure that the proper positioning for restricted flow to a ruptured cold leg -ensuring is maintained. The verification of proper position of a throttle/runout valve can be accomplished by confirming the seals and lock have not been altered since the last performance of the flow balance test. Restricting the flow to a ruptured cold leg ensures and that the other cold legs receive at least the required minimum flow. This Surveillance is not required for-~~

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SURVEILLANCE
REQUIREMENTS
(continued)

~~plants with flow limiting orifices.~~ The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

Periodic inspections of the containment recirculation sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

Opening the containment recirculation sump access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered. (Ref. 9)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR, Sections 6.3 and 7.3.
4. FSAR, Chapter 15, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.
7. BTP EICSB-18, Application of the Single Failure Criteria to Manually-Controlled Electrically-Operated Valves
8. ASME/ANSI OM-1987, Operational Maintenance of Nuclear Power Plants, including OM-a-1988 addenda, Part 6, "Inservice Testing of Pumps in Light Water Reactor Power Plants," and part 10, "Inservice Testing of Valves in Light Water Reactor Power Plants"
9. NRC Letter to PG&E, EA 89-241, April 5, 1990, CHRON 148598

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2, and subsequently transferring RHR pump suction to the containment recirculation sump.

APPLICABLE
SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable core reactivity and the lower heat removal conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuations are not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA. (Ref 1)

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation.

The ECCS trains satisfy Criterion 3 of the NRC Policy Statement. 10 CFR 50.36(c)(2)(ii)

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains (as defined for MODE 4) is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

The LCO is modified by a Note that allows a RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned.

BASES

LCO (continued)

(remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment recirculation sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS charging and RHR pumps and their respective supply headers to each of the four cold legs. injection nozzles. In the long term, this flow path may be switched to take its supply from the containment recirculation sump and to deliver its flow to the RCS hot and cold legs.

This LCO is modified by a Note that allows an RHR train to be considered OPERABLE during system alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS high head and low head train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

ACTIONS (continued)

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat

BASES

exchangers. The Completion Time of "immediately" to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

Opening the containment recirculation sump access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered. (Ref. 9)

B.1

With no ECCS centrifugal charging high-head subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS centrifugal charging high-head subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE
REQUIREMENTSSR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply. ~~This SR is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being~~

BASES

~~manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4, if necessary.~~

REFERENCES

1. Abnormal Response Guideline, ARG-2, Rev. 0, Feb. 28, 1992

Note: The applicable references from Bases 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

~~The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling, and to the ECCS and the Containment Spray System during accident conditions.~~

~~The RWST supplies both trains of the ECCS and the Containment Spray System through separate, redundant supply headers during the injection phase of a loss of coolant accident (LOCA) recovery. A motor operated isolation valve is provided in each header to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment recirculation sump following receipt of the RWST Low Low (Level 1) signal. Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.~~

~~The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS volume control tank (VCT) to the RWST through the use of isolation valves. Each set of isolation valves is interlocked so that the VCT isolation valves will begin to close once the RWST isolation valves are fully open. Since the VCT is under pressure, the preferred pump suction will be from the VCT until the tank is isolated. This will result in a delay in obtaining the RWST borated water. The effects of this delay are discussed in the Applicable Safety Analyses section of these Bases.~~

~~During normal operation in MODES 1, 2, and 3, the safety injection (SI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.~~

~~The ECCS and Containment Spray System pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at or near shutoff head conditions.~~

~~When the suction for the ECCS and Containment Spray System pumps is transferred to the containment recirculation sump, the RWST flow paths must be isolated to prevent a release of the containment recirculation sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the~~

BASES

~~ECCS pumps.~~

~~This LCO ensures that:~~

- ~~a. The RWST contains sufficient borated water to support the ECCS during the injection phase;-~~
- ~~b. Sufficient water volume exists in the containment recirculation sump to support continued operation of the ECCS and Containment Spray System pumps at the time of transfer to the recirculation mode of cooling; and~~
- ~~c. The reactor remains subcritical following a LOCA.~~

~~Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.~~

~~The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions (boration flow path); to the refueling cavity during refueling, and to the ECCS and the Containment Spray (CS) System during accident conditions.~~

~~The RWST supplies both trains of the ECCS through one header and both trains of the CS System through a separate supply header during the injection phase of a loss of coolant accident (LOCA) recovery. Motor-operated isolation valves in each sub-system header isolate the RWST from the ECCS and from the CS System once the RWST is no longer supplying flow to these systems.~~

~~Use of a single RWST to supply both trains of the ECCS and CS Systems is acceptable since the RWST is a passive component, and a passive failure is not assumed to occur coincidentally with a Design Basis Accident (DBA).~~

~~During normal plant operation in MODES 1, 2, and 3, the Safety Injection (SI) and Residual Heat Removal (RHR) pumps are aligned to take suction from the RWST. The Centrifugal Charging Pumps (CCPs) operate during normal plant operation with their suction aligned to the Volume~~

BASES

BACKGROUND (continued)

Control Tank (VCT). The switchover from normal operation to the injection phase of ECCS operation requires auto-transfer of the CCP suction from the CVCS VCT to the RWST. The CS pumps suction is aligned to the RWST with closed motor operated discharge valves which open on a CS signal. When the suction for the RHR pumps is transferred to the containment recirculation sump, the RWST must be isolated from ECCS and CS system. The isolation prevents flow of containment recirculation sump water into the RWST. Flow of containment water into the RWST could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the RHR pumps due to loss of containment recirculation sump inventory.

The reactivity control systems are available to the operators to ensure that negative reactivity is available during each mode of plant operation. This system is not an automatic accident mitigation system, but is used under operator control if needed to increase the Reactor Coolant System (RCS) boration concentration. The sources of borated water are the boric acid storage tanks in the CVCS and the RWST. The RWST source of borated water is available as an alternate source to the boric acid storage tanks. RWST water can be used in the event of abnormal conditions, including single active failure events that may impair the function of the boric acid storage tank source of borated water of the CVCS. The boration subsystem provides the means to meet one of the functional requirements of the CVCS, i.e., to control the neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN (SDM).

The LCO ensures that:

- a. The RWST contains a sufficient volume at an acceptable boron concentration and temperature to support the ECCS and CS systems during the injection phase;
- b. Sufficient water volume exists in the containment recirculation sump to support continued operation of the ECCS System pumps at the time of transfer to the recirculation mode of cooling;
- c. The reactor remains subcritical following a LOCA

APPLICABLE
SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and CS Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and

BASES

APPLICABLE SAFETY
ANALYSES (continued)

applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating"; B 3.5.3, "ECCS - Shutdown"; and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

~~The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume.~~

~~The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered.~~

~~The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The importance of its value is small for units with a boron injection tank (BIT) with a high boron concentration. For units with no BIT or reduced BIT boron requirements, the minimum boron concentration limit is an important assumption in ensuring the required shutdown capability. The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a non-limiting event and the results are very insensitive to boron concentrations. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a feedline break is consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically non-limiting.~~

~~The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as [27] seconds, with offsite power available, or [37] seconds without offsite power. This response time includes [2] seconds for electronics delay, a [15] second stroke time for the RWST valves, and a [10] second stroke time for the VCT valves. Plants with a BIT need not be concerned with the delay since the BIT will supply highly borated water prior to RWST switchover, provided the BIT is between the pumps and the core.~~

BASES

APPLICABLE SAFETY
ANALYSES (continued)

For a large break LOCA analysis, the minimum water volume limit [~~466,200~~] ~~428,237~~ ~~400,000~~ gallons (~~91%~~) (~~81.5%~~) and the lower boron concentration limit of [~~2000~~] ~~2400~~ ~~2300~~ ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of [~~2200~~] ~~2600~~ (2500) ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.

In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of [~~35~~] ~~35~~°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of [~~100~~] ~~120~~°F is used in the small break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA and higher containment pressures due to reduced containment spray cooling capacity. For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.

Any event that results in SI initiation, including inadvertent ECCS actuation, results in delivery of RWST water to the RCS. However, the events for which the RWST parameters provide mitigation or are limiting are large break LOCA and steam line break. Feedwater line break and steam generator tube rupture (SGTR) also involve SI but the RWST parameters are less significant to the analysis results. RWST boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a non-limiting event and the results are very insensitive to boron concentrations. The effect of these RWST parameters on large break LOCA, main steam line break, feedwater line break, and SGTR are discussed below:

LOCAVolume

Insufficient water in the RWST could result in insufficient borated water inventory in the containment

BASES

APPLICABLE SAFETY
ANALYSES (continued)

recirculation sump when the transfer to the recirculation phase occurs. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is less than the total volume contained since, due to the design of the tank, the ECCS suction nozzle elevation is above the bottom of the tank, so more water can be contained than can be delivered. The contained water volume limit includes an allowance for water not usable because of tank discharge location or other physical characteristics.

Boration

During accident conditions, the RWST provides a source of borated water to the ECCS and CS System pumps. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment. The minimum boron concentration limit ensures that the spray and the containment recirculation sump solutions, after mixing with the sodium hydroxide from the spray additive tank, will not exceed the maximum pH values. The maximum boron concentration limit ensures that the containment recirculation sump solution will not be less than the minimum pH requirement. The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Diablo Canyon FSAR Update. These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

For a large-break LOCA analysis, the RWST minimum contained water volume of 400,000 gallons (81.5% indicated level, uncorrected for uncertainty) and the lower boron concentration limit of 2300 ppm are used to compute the post-LOCA sump boron concentration necessary to assure subcriticality. The large-break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2500 ppm is used to determine the maximum allowable time to initiate hot leg recirculation following a LOCA. The purpose of initiating hot leg recirculation is to avoid boron precipitation in the core following the accident when the break is in the cold leg.

The use of minimum containment backpressure in the LOCA analysis results in a conservative calculation of Peak Clad Temperature (PCT). The basis for this conclusion is the effect that the containment pressure has on the core reflood rate. A lower containment pressure has the effect of reducing the density of the steam exiting the break.

BASES

APPLICABLE SAFETY
ANALYSES (continued)

which increases the differential pressure provided by the downcomer head (this phenomena is sometimes referred to as steam binding). Thus, a higher downcomer mixture level is required to maintain the same reflood rate as before. The additional time required to establish the downcomer head translates into a reduction in the reflood rate in the core. When the downcomer has completely filled, the equilibrium reflood rate for the low containment pressure case would be less than that calculated for a high containment pressure case. This reduction in reflood rate results in a reduction in heat transfer and ultimately an increase in the calculated POT. Thus, the regulations require that a low containment pressure be calculated in the large-break LOCA analysis.

In the ECCS analysis, the CS temperature is assumed to be equal to the RWST minimum temperature limit of 35°F. If the minimum temperature limit is violated, the CS containment spray further reduces containment pressure, which decreases the core reflood as explained in the preceding paragraph. For the containment response following a MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

Steam Line and Feedwater Line BreaksVolume

RWST volume is not an explicit assumption in other than LOCA events since the required volume for those events is much less than that required by LOCA.

Boration

The minimum RWST solution boron concentration is an explicit assumption in the MSLB analysis to ensure the required shutdown capability. Since DCPD no longer uses the boron injection tank, the minimum boron concentration limit is an important assumption in ensuring the required shutdown capability. For the containment response following an MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

Feedwater line break results in high temperature/high pressure in the RCS. There is very little RWST water injected due to the high pressure. Also, the analysis results are not affected by the negative reactivity provided by RWST water. Therefore, RWST boron concentration is not a consideration for the feedwater line break.

BASES

APPLICABLE SAFETY

ANALYSES ~~Steam Generator Tube Rupture (SGTR)~~

(continued)

Volume

The RWST volume needed in response to a SGTR is not an explicit assumption since the required volume is much less than that required by a LOCA.

Boration

Borated RWST water will be injected into the RCS for a SGTR event. The insertion of the control rods and the negative reactivity provided by the injected RWST solution provides sufficient SDM during the initial recovery operations. One of the initial operator recovery actions for this event is to equalize the RCS pressure and the faulted steam generator pressure to minimize or stop the primary-to-secondary tube rupture flow and terminate safety injection. Further RCS boration will be initiated by the operator by manual makeup to the RCS.

The RWST satisfies Criterion 3 of the NRC Policy Statement 10 CFR 50.36(c)(2)(ii).

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment recirculation sump to support ECCS and ~~Containment Spray System~~ pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and ~~CS Containment Spray System~~ OPERABILITY requirements. Since both the ECCS and the ~~CS Containment Spray System~~ must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual

BASES

APPLICABILITY
(continued)

Heat Removal (RHR) and Coolant Circulation—Low Water Level."

ACTIONS

A.1

With RWST boron concentration or borated water temperature* not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the ~~CS~~ Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

DGPP GTS does not have an upper limit for RWST borated water temperature. An upper limit would typically be about 100°F. The coastal weather at the DGPP site is moderated by the Pacific Ocean and historically does not exceed 100°F. A requirement for a high temperature limit would therefore not be of value.

*
§

The requirement for RWST temperature is to be greater than or equal to the minimum required temperature. The expression "within the required limits" applied to RWST temperature is satisfied when the temperature is greater than or equal to the minimum.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the ~~CS~~ Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which

BASES

ACTIONS (continued)

the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains and that borated water volume can be restored more rapidly than boron concentration or temperature.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits above the minimum assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either the limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of or above the minimum temperature for the RWST. With ambient air temperature within the band, above the minimum temperature the RWST temperature should not exceed the limit.

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for ECCS injection and CS pump operation and to support continued ECCS on recirculation. Since the RWST volume is normally stable and the contained volume required is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

The required RWST water volume of 95% (428,237 gallons, 91% plus 4% measurement uncertainty) is surveilled by the control board indication.

BASES

SURVEILLANCE
REQUIREMENTS (continued)

The analysis assumed 400,000 gallons (81.5% of indicated range) is used in the TS Surveillance and is shown on the control board indicators. Adjustments to the analysis parameters for instrument inaccuracies or other reasons are applied to determine the acceptance criteria used in the plant surveillance procedures. These adjustments assure the assumed analyses parameters are maintained.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR, Chapter 6 and Chapter 15.
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

This LCO is applicable because only to those units that utilize the centrifugal charging pumps CCPs are utilized for high head safety injection (SI). The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the CCP pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during SI.

APPLICABLE
SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps while the inadvertent SI and the SGTR analyses establish the maximum flow for the ECCS pumps. The centrifugal charging pumps CCPs are also credited in the small break LOCA analysis. This analysis establishes the flow and discharge head at the design point for the CCPs. The steam generator tube rupture and main steam line break event analyses also credit the CCPs but are not limiting in their design requirements. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

The ECCS flow balance assumes RCP seal injection is limited to 40 gpm with the flow control valve fully open.

This LCO ensures that total seal injection flow of ≤ 40 gpm, with CCP discharge header RCS pressure $\geq [2480]$ 2215 psig and ≥ 2255 psig and charging flow control valve full open, will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the CCPs will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory.

Seal injection flow satisfies Criterion 2 of the NRC Policy Statement 10 CFR 50.36(c)(2)(ii)

(continued)

BASES

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the injection points cold legs (Ref. 2 1). This is accomplished by limiting the line resistance in the RCP seal injection lines to a value consistent with the assumptions in the accident analysis.

The 40 gpm identified in the LCO is not strictly an absolute flow limit, but rather a flow limit through the RCP seal injection line that is assumed in the accident analyses initial conditions when the ECCS systems are aligned in the injection mode following a LOCA. This flow value correlates to a line resistance in the seal injection flow path that is used in the accident analyses ECCS performance. Thus, the line resistance is the parameter which is controlled to ensure that the ECCS alignment is maintained consistent with the accident analysis assumptions. Charging flow control valve, FCV-128 full open is a test condition and is not indicative of normal operation. Consequently, during normal plant operation, it is possible to have the indicated total seal injection flow greater than 40 gpm while still being within the LCO because during normal plant operation, the ECCS system is not in post accident alignment. Based on flow line resistance with the CCPs aligned for safety injection.

In order to establish the proper flow line resistance, the seal injection flow path differential pressure and flow are measured. The line resistance is then determined with the RCS pressure within normal limits and the CCP flow control valve fully open, a pressure and flow must be known. The flow line resistance is determined by assuming that the RCS pressure is at normal operating pressure and that the CCP discharge pressure is in this LCO. The CCP discharge header pressure remains essentially constant through all the applicable MODES of this LCO. A reduction in RCS pressure, with no concurrent decrease in CCP discharge header pressure, would result in more flow being diverted to discharged through the RCP seal injection line than at normal RCS operating pressure. The RCP seal injection valve settings established at the prescribed CCP discharge header RCS pressure result in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the charging flow control valve (charging flow for four loop units and air operated seal injection for three loop units) being full open, is required since consistent with the air operated valve is designed assumed to fail open for the accident condition.

(continued)

BASES

LCO
(continued)

With the discharge RCS pressure and control valve position as specified by the LCO, a flow limit is established which assures that the seal injection line resistance is consistent with the analysis assumptions. ~~a limit is established. It is this line limit resistance that is used in the accident analyses. This limit assures that when the RCS depressurizes following a LOCA and the flow to the pump seals increases, the resulting flow to the seals will be less than the limit assumed in the accident analysis.~~

~~The limit on seal injection flow, combined with the CCP discharge header pressure limits and an open wide condition of the charging flow control valve, must be met to render the ECCS OPERABLE. If these conditions are not met, the ECCS flow will not be as assumed in the accident analyses.~~

APPLICABILITY

In MODES 1, 2, and 3, the seal injection flow limit is dictated by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in ~~this MODE 4~~. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTIONS

A.1 and A.2

With the seal injection flow exceeding its limit, the amount of charging flow available for ECCS injection to the RCS may be reduced. Under this Condition, action must be taken to restore the seal injection flow to below its limit. Required Action A.1 ensures that within 4 hours the remaining available ECCS charging flow (without assuming an additional failure) is $\geq 100\%$ of the assumed post-LOCA charging flow. 100% flow capability may be verified by assuring both CCPs are OPERABLE. Required Action A.2 then allows the operator has 4 72 hours from the time the flow is known to be above the limit but still allowing 100% of the assumed post-LOCA ECCS charging flow, to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is conservative consistent with respect to the Completion Times for ECCS in 3.5.2 ACTION A.1 of other ECCS LCOs; it is based on operating experience and is sufficient for taking corrective actions by operations personnel.

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE
REQUIREMENTS

SR 3.5.5.1

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a flow ~~within~~ below the limit ensures ~~that~~ proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve Surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.

As noted, the Surveillance is ~~not required to be performed until completed within 4 hours~~ after the RCS pressure has stabilized within a ~~± 20 psig range of normal operating the specified pressure limits~~. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely.

REFERENCES

1. FSAR, Chapter 6 and Chapter 15.
 2. 10 CFR 50.46.
-
-

Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(2 Pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431
NUREG-1431 Section 3.5

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE
NUMBER

JUSTIFICATION

- | | |
|-------|---|
| 3.5-1 | This change replaces reference to the "pressurizer pressure" with a reference to the "RCS pressure" in the APPLICABILITY, Required Action C.2, and SR 3.5.1.5. Required ACTION C.2 requires reducing pressurizer pressure to less than 1000 psig. However, pressurizer pressure instrumentation does not have the range to read that pressure. Consequently, RCS pressure instrumentation is used. For the purposes of this LCO, the use of RCS pressure is equivalent. This is consistent with Industry Traveler 117. |
| 3.5-2 | Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 6B). |
| 3.5-3 | This change adds the word "mechanical" with regard to throttle valve position stop, consistent with the CTS. These valves have mechanical stops that maintain the valves in position for proper ECCS performance. |
| 3.5-4 | This change increases the RCP seal injection flow Completion Time from 4 to 72 hours, with a new added verification that at least 100 percent of the assumed charging flow remains available. The Bases for seal injection flow relate the limit to ensuring adequate charging flow during post-LOCA injection. The revised ACTIONS continue to assure this basis is adequately addressed by providing an ECCS-like Required Action. ITS 3.5.2 allows a 72 hour Completion Time for 1 or more ECCS subsystems inoperable if at least 100 percent of the assumed ECCS flow is available. The seal injection flow ACTIONS have been modified so that if the remaining charging flow (with some inoperability in the charging system) is greater than or equal to 100 percent of the assumed post-LOCA charging flow, 72 hours is allowed to restore OPERABILITY. This change is consistent with industry Traveler WOG-84. |
| 3.5-5 | This change deleted reference to CCP discharge header pressure from the LCO and ACTION A to reflect CTS [3.4.6.2.]. A description is added to the Bases which provides the methodology for adjusting the seal injection throttle valves consistent with plant-specific analyses. |

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431
NUREG-1431 Section 3.5

**CHANGE
NUMBER**

JUSTIFICATION

- | | |
|-------|---|
| 3.5-6 | SR 3.5.3.1 Note (An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being realigned to the ECCS mode of operation) is moved to the LCO per industry Traveler TSTF-90, Rev. 1.

The Note is more appropriately placed in the LCO because it defines the intended capability of the ECCS equipment. |
| 3.5-7 | Not used. |
| 3.5-8 | Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). |
| 3.5-9 | Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B). |

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(8 Pages)

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-01	ACTION A.1 is revised [new ACTION A.2 is added , and Table 3.7.1-1 is revised] to account for operation with inoperable MSSVs and resetting the power range neutron flux high trip setpoints with inopreable MSSVs. The Compelction Time to reset the power range neutron flux high trip setpoints is extended to 72 hours. [The table is revised based on NSAL 94-001 and deletes the reference to five OPERABLE MSSVs.]	Yes	Yes	Yes	YES
3.7-02	The CTS Applicability of MODES 1, 2, and 3 is being retained in ITS 3.7.2, MSIVs, and ITS 3.7.3, MFIVs.	No, DCPD is adopting ITS.	No, CPSES is adopting ITS.	Yes	YES
3.7-03	SR 3.7.3.1 is divided into two surveillances since both the stroke time and the Frequency requirements are different at DCPD for the feedwater regulation/bypass valves and the feedwater isolation valve.	Yes, per LA 77/76	No	No	No
3.7-04	Requirements involving reliance on the SG heat removal system for heat removal in MODE 4 would be deleted.	No, DCPD is adopting ITS.	Yes	Yes	Yes
3.7-05	Required Action B.1 and new C.1 are revised to state that restoration of "all but one" and "all but two" [ADV] lines is required which will effectively exit the respective Required Action.	Yes	Yes	No, refer to 3.7-19.	No, refer to 3.7-19.
3.7-06	The Condition and Required Action for two or more inoperable [ADV] lines is limited to two [ADV] lines and the Completion Time is revised from 24 to 72 hours per the current licensing basis. A new Condition C is added.	Yes	Yes	No, not part of CTS.	No, not part of CTS.
3.7-07	Revised Conditions A and C to be consistent with CTS. The ITS as written would have allowed the OPERABLE EES train to remain in standby during movement of irradiated fuel.	No, not part of CTS.	No, not part of CTS.	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-08	SR 3.7.5.1 is revised to add a Note consistent with the CTS and the plant specific design. The verification of flow control valve position is deferred until conditions are appropriate.	No, AFW valves have a correct position.	Yes	Yes	Yes
3.7-09	New Conditions F, G, and H and the surveillance requirement associated with the FWST AFW pump supply are relocated from the CTS on AFW supply and included in the DCPD AFW specification for completeness.	Yes	No	No	No
3.7-10	The specification description, the LCO, the ACTION requirements and the Surveillance are revised to incorporate the DCPD plant specific requirement for OPERABLE AFW supply sources via the CST and the FWST per the current licensing basis.	Yes	No	No	No
3.7-11	The Required Actions for CPSES feedwater isolation and associated bypass valves inoperable are revised consistent with the current licensing basis for a Completion Time of 4 hours and to credit the MFRVs (feedwater control valves (FCVs)) and associated bypass valves for a Completion Time of 72 hours. A new SR is added for the FCVs and associated bypass valves.	No	Yes	No	No
3.7-12	WOG-83 revised Condition A and Table 3.7-1 to account for plants that credit the Power Range High Neutron Flux trip function when MTC is positive (See 3.7-01 above). The wording of the traveler has been modified for CPSES to account for plant specific differences.	No	Yes	No	No
3.7-13	Note 1. under Action Required A.1 is deleted. The DCPD emergency diesel generators have self contained cooling systems that do not rely upon an external source of cooling water.	Yes	No	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-14	The Note for SR 3.7.8.1 is deleted since the DCPD ASW system supplies only the CCW heat exchanger and no other individual components.	Yes	No	No	No
3.7-15	SR 3.7.8.1 is revised to include a DCPD specific requirement to verify the availability of power and air so that the valves can be placed in their correct positions as described in the Bases.	Yes	No	No	No
3.7-16	SR 3.7.8.2 is revised to include only power operated valves since there are no automatically operated valves in the DCPD ASW system. The surveillance is revised to require movement of power operated valves to demonstrate the ability to reconfigure the ASW system as described in the FSAR and the ITS Bases.	Yes	No	No	No
3.7-17	The DCPD UHS specification is revised to reflect the fact that for the system to perform its intended function, it is temperature limited. The LCO is revised to note that the specification limits the temperature to less than or equal to 64°F. SR numbers are revised accordingly and the remaining surveillances are deleted as being not applicable.	Yes	No	No	No
3.7-18	The DCPD specific CTS surveillance for leakage testing of ABVS dampers M2A and M2B is retained.	Yes	No	No	No
3.7-19	Required Action B.1 is revised to state that restoration of "all but" one [ASD] line is required, which will effectively exit Required Action B.1 and re-enter Required Action A.1.	No, refer to change 3.7-5 and 3.7-6.	No, refer to change 3.7-5 and 3.7-6.	Yes	Yes
3.7-20	A Callaway specific Condition is added to address the inoperability of one of the Essential Service Water (ESW) supplies to the turbine-driven AFW pump.	No	No	No	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-21	The ACTIONS and SR 3.7.12.1 are revised to include the DCPD specific CTS ACTION requirement for an inoperable HEPA filter and/or charcoal absorber and to provide the appropriate charcoal absorber monthly drying time for the single common charcoal absorber.	Yes	No	No	No
3.7-22	SR 3.7.12.3 is revised for DCPD to describe the expected actions upon an actuation of the ABVS.	Yes	No	No	No
3.7-23	ACTION C and SR 3.7.12.4 are deleted since the DCPD ABVS was not designed to maintain a specific negative pressure.	Yes	No	No	No
3.7-24	SR 3.7.12.5 is deleted since there are no DCPD ABVS bypass dampers and the system automatic dampers are tested by SR 3.7.12.3.	Yes	No	No	No
3.7-25	Based on the CTS, a Note is added to [SR 3.7.3.1, 3.7.3.2, 3.7.4.1 and 3.7.4.2] to indicate that demonstration of valve OPERABILITY is only required to be performed for entry into (and continued operation in) MODES 1 and 2. This Note states that the SR is only required to be performed in MODES 1 and 2. This would allow entry into MODE 3 for the purpose of testing the valves.	No, note is not part of CTS.	No, note is not part of CTS.	Yes	Yes
3.7-26	Condition D is deleted to reflect the CPSES plant specific design of primary FIVs and associated bypass valves and isolation backup via the in series FCVs and associated bypass valves.	No	Yes	No	No
3.7-27	A Note is added to DCPD Table 3.7.1-2 under Lift Setting that specifies that the lift point of the lowest set safety is +3% and -2%.	Yes (per LA 108/107)	No	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-28	Revise [ADV] Frequency from 18 months to "in accordance with Inservice Test Program."	No, CTS is 18 months.	Yes	Yes	Yes
3.7-29	Revise AFW pump testing to be "In accordance with Inservice Test Program."	Yes	Yes	Yes	Yes
3.7-30	LCO 3.7.8 and ACTIONs are revised for CPSES to incorporate requirements for two units with station service water system cross connections.	No, covered by ECG per GL 91-13 response.	Yes	No, single unit plant.	No, single unit plant.
3.7-31	SR 3.7.8.2 is replaced with the current CPSES specific surveillance of the cross connections between units. The CPSES design has no automatic valves as per this SR in the ITS.	No, refer to 3.7-15 and 3.7-16.	Yes	No	No
3.7-32	CONDITION A for CPSES is changed to "SSI level less than required" and SR 3.7.9.3 and 3.7.9.4 are deleted.	No	Yes	No	No
3.7-33	The requirement to verify a make-up flow rate during the tests demonstrating the capability to maintain control room differential pressure above atmospheric pressure would be deleted.	Yes, per CTS.	No, retained CTS requirement.	Yes	Yes
3.7-34	In accordance with Traveler WOG-64, the Completion Time for closing one inoperable MSIV is extended to 72 hours; and separate Required Actions are included for either one MSIV inoperable or two or more MSIVs inoperable in MODES 2 and 3.	No, adopting 8 hour AOT from STS.	Yes	Yes	Yes
3.7-35	SR 3.7.10.3 is revised to reflect DCPD specific plant configuration and CTS required testing.	Yes	No	No	No
3.7-36	Required Actions D and E are revised for CPSES for two trains inoperable where at least 100% of the required heat removal capacity is available.	No	Yes	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-37	Modifies LCO 3.7.2 Condition A and adds new Condition B and C to be consistent with the CPSES CTS.	No	Yes	No	No
3.7-38	This proposed change deletes reference to a specific flowrate for conducting the negative pressure test per the CPSES CTS.	No, see CN 3.7-49.	Yes	No, see CN 3.7-49.	No, see CN 3.7-49.
3.7-39	SR 3.7.12.6 is added to verify the shutdown of the non-ESF fans to prevent bypass of the ESF Filtration units (CPSES specific).	No	Yes	No	No
3.7-40	Not used.	N/A	N/A	N/A	N/A
3.7-41	The Main Feedwater Regulating and associated Bypass Valves are deleted from the ITS per current licensing basis.	No, CTS includes MFRVs.	No, refer to 3.7-11.	Yes	Yes
3.7-42	Add DCPD specific note that states that 3.0.3 is not applicable to the fuel handling building ventilation system during fuel movement since fuel movement is independent of reactor operation.	Yes	No	No	No
3.7-43	ACTION A of ITS 3.7.13 is revised and ACTIONS C, E and F.1 of ITS 3.7.13 are not used per the DCPD CTS.	Yes	No	No	No
3.7-44	This change would revise ITS 3.7.13 to add a new Note to the Applicability and change the Conditions, Required Actions, and SRs to conform to the design of the Emergency Exhaust System.	No, fuel building ventilation not required for post LOCA leakage.	No, CTS does not require this specification.	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-45	ITS 3.7.15 is revised to be CPSES specific to address the two spent fuel pools and the in-containment storage racks.	No	Yes	No	No
3.7-46	Revised to delete "irradiated fuel assemblies seated in" since accident analysis assumes fuel assembly lying on top of the fuel storage racks.	No, ITS is consistent with CTS.	Yes	No, ITS is consistent with CTS.	Yes
3.7-47	This change adds TS 3.7.19, a safety chilled water system which is in the CPSES CTS.	No	Yes	No	No
3.7-48	This change adds TS 3.7.20, an UPS HVAC system which is in the CPSES CTS.	No	Yes	No	No
3.7-49	The requirement to verify a make-up flow rate during the tests demonstrating the capability to maintain [fuel handling] building differential pressure below atmospheric pressure would be deleted. The current licensing basis of the plant is to be able to maintain a negative pressure [in the fuel handling building] with respect to the outside atmosphere.	Yes	No, see CN 3.7-38.	Yes	Yes
3.7-50	The CTS DCPD specific ADV surveillance that verifies the back-up air bottle pressure once per 24 hours is retained.	Yes	No	No	No
3.7-51	A new spent fuel pool storage specification is created for Region 1 fuel storage due to unique storage requirements at DCPD.	Yes	No	No	No
3.7-52	ITS 3.7.14 is not used due to the mild coastal environment in which the plant is located.	Yes	No	No	No
3.7-53	ITS 3.7.16 for DCPD is revised to be consistent with the current licensing basis and CTS.	Yes	No	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-54	The LCO, Required Actions, and Surveillances are revised per the DCPD specific CTS to incorporate Region 2 fuel storage requirements.	Yes, per LA 116/114.	No	No	No
3.7-55	NUREG-1431 Specification 3.7.14 is not used since an equivalent safety grade system does not exist. Therefore, the deletion is per the current licensing basis.	Yes	No	No	No
3.7-56	This change creates a new SR for the MSIVs [and MFIVs] to distinguish between the IST and the automatic actuation testing of these isolation valves. The SR allows credit for an actual actuation, if one occurs, to satisfy the surveillance requirements. Although SRs 3.7.2.2 and 3.7.3.2 are new SRs, they may be performed in conjunction with SRs 3.7.2.1 and 3.7.3.1. Therefore, the Note allowing testing to be performed on MODE 3 is also needed for these new SRs.	Yes	Yes	Yes	Yes
3.7-57	This change establishes appropriate Required Actions and Completion Times for ventilation system pressure envelope degradation.	{No, retained CTS.}	Yes	Yes	Yes

JLS Conversion to Improved Technical Specifications

Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 3/4.6 - Containment Systems

ITS 3.6 - Containment Systems



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.6

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- ENCLOSURE 4 - NO SIGNIFICANT HAZARDS CONSIDERATIONS
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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(5 Pages)
CONVERSION TABLE SORTED BY IMPROVED TS	(4 Pages)
METHODOLOGY	(2 Pages)

CROSS-REFERENCE TABLE FOR 3/4.6
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.6.1.1	LCO		01-01-LG	3.6.1	LCO		
3.6.1.1	APP			3.6.1	APP		
3.6.1.1	ACTION		01-01-LG	3.6.1	ACTION		
4.6.1.1	SR			3.6.1.1	SR		3.6-1
4.6.1.1	SR	b	01-05-A	3.6.2.1	SR		3.6-1
4.6.1.1	SR	a	01-02-A 01-04-LS1 01-06-LS19	3.6.3.3	SR		3.6-5
4.6.1.1	SR	a	01-02-A 01-04-LS1 01-06-LS19	3.6.3.4	SR		3.6-5
4.6.1.1	SR	a	01-02-A 01-03-A 01-04-LS1 01-06-LS19	3.6.3	ACTION	A.1 A.2	3.6-11
3.6.1.2	LCO		02-01-A	5.5.16			
3.6.1.2	APP		02-01-A	3.6.1	APP		
3.6.1.2	ACTION		02-01-A 02-02-A 02-06-A	3.6.1	ACTION		
3.6.1.2	ACTION		02-02-A 02-06-A	5.5.16			
4.6.1.2	SR			5.5.16			
3.6.1.3	LCO		03-01-LG	3.6.2	LCO		
3.6.1.3	APP			3.6.2	APP		
3.6.1.3	ACTION		03-02-A 03-09-LS7 03-13-A	3.6.2	ACTION	NOTES	
3.6.1.3	ACTION	a.1, a.2	03-03-LS3 03-04-M 03-05-LS4 03-06-LS5 03-08-LS6	3.6.2	ACTION	A, B	
3.6.1.3	ACTION	a.3		3.6.2	ACTION	D	
3.6.1.3	ACTION	a.4	03-12-A			Not Used	
3.6.1.3	ACTION	b	03-07-M	3.6.2	ACTION	C, D	
4.6.1.3	SR	a		3.6.2.1	SR		3.6-1

CROSS-REFERENCE TABLE FOR 3/4.6
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
4.6.1.3	SR	b	03-10-LS8	3.6.2.2	SR		3.6-2
3.6.1.4	LCO			3.6.4	LCO		
3.6.1.4	APP			3.6.4	APP		
3.6.1.4	ACTION			3.6.4	ACTION		3.6-8
4.6.1.4	SR			3.6.4.1	SR		
3.6.1.5	LCO			3.6.5	LCO		
3.6.1.5	APP			3.6.5	APP		
3.6.1.5	ACTION			3.6.5	ACTION		
4.6.1.5	SR		05-01-LG	3.6.5.1	SR		
3.6.1.6	LCO		06-02-A	3.6.1	LCO		
3.6.1.6	APP			3.6.1	APP		
3.6.1.6	ACTION		06-04-M	3.6.1	CONDITION		
4.6.1.6.1	SR		06-02-A	3.6.1.1	SR		3.6-1
4.6.1.6.2	SR		06-03-TR2			Not Used	
3.6.1.7	LCO		07-01-A	3.6.3	LCO		
3.6.1.7	LCO		07-04-R	3.6.3	ACTION	Notes	3.6-17
3.6.1.7	APP			3.6.3	APP		
3.6.1.7	ACTION		03-13-A 07-04-R 07-05-A 11-02-A 11-12-A	3.6.3	ACTION	D	
3.6.1.7		New	01-04-LS1 07-02-LS9	3.6.3	ACTION	D	
4.6.1.7.1	SR		07-03-A	3.6.3.2	SR		3.6-17
4.6.1.7.2	SR		07-04-R			Not Used	
4.6.1.7.3	SR			3.6.3.10	SR		
3.6.2.1	LCO		08-01-LG 08-04-A	3.6.6	LCO		3.6-14

CROSS-REFERENCE TABLE FOR 3/4.6
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.6.2.1	APP			3.6.6	APP		
3.6.2.1	ACTION		08-02-A 08-11-LS2	3.6.6	ACTION	A, B	
4.6.2.1	SR	a		3.6.6.1	SR		
4.6.2.1	SR	b	08-08-LG	3.6.6.4	SR		
4.6.2.1	SR	c.1		3.6.6.5	SR		
4.6.2.1	SR	c.2		3.6.6.6	SR		
4.6.2.1	SR	d	08-06-LG	3.6.6.8	SR		
3.6.2.2	LCO	a	09-01-A	3.6.7	LCO		
3.6.2.2	LCO	b	09-02-LG			Not Used	
3.6.2.2	APP			3.6.7	APP		
3.6.2.2	ACTION		09-03-A	3.6.7	ACTION	A, B	
4.6.2.2	SR	a		3.6.7.1	SR		
4.6.2.2	SR	b.1		3.6.7.2	SR		3.6-10
4.6.2.2	SR	b.2		3.6.7.3	SR		
4.6.2.2	SR	c	09-04-A 09-05-TR1	3.6.7.4	SR		
4.6.2.2	SR	d	09-07-M	3.6.7.5	SR		
3.6.2.3	LCO		08-04-A	3.6.6	LCO		3.6-14
3.6.2.3	APP			3.6.6	APP		
3.6.2.3	ACTION	a	08-11-LS2	3.6.6	ACTION	C	3.6-14
3.6.2.3	ACTION	b		3.6.6	ACTION	D, E	3.6-14
3.6.2.3	ACTION	New	08-10-A	3.6.6	ACTION	F	
4.6.2.3	SR	a.1		3.6.6.2	SR		
4.6.2.3	SR	a.2	10-03-LG	3.6.6.3	SR		
4.6.2.3	SR	a.3		3.6.6.9	SR		
4.6.2.3	SR	b	09-05-TR1	3.6.6.7	SR		3.6-14
3.6.3	LCO		11-01-LS13 11-11-A	3.6.3	LCO		3.6-3
3.6.3	APP			3.6.3	APP		

CROSS-REFERENCE TABLE FOR 3/4.6
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.6.3	ACTION		03-13-A 11-02-A 11-03-A	3.6.3	ACTION	NOTES	3.6-17
3.6.3	ACTION	a	11-16-A			Not Used	
3.6.3	ACTION	b,c	01-03-A 11-12-A	3.6.3	ACTION	A	
3.6.3	ACTION	d		3.6.3	ACTION	E	
3.6.3		New	11-04-A	3.6.3	ACTION	B	
3.6.3		New	11-05-LS14	3.6.3	ACTION	C	3.6-4 3.6-11
4.6.3.1	SR		11-06-TR3			Not Used	
4.6.3.2	SR		11-07-LG 11-14-A 11-08-TR1 11-09-A	3.6.3.8	SR		
4.6.3.2	SR	b,c	11-07-LG			Not Used	
4.6.3.3	SR		11-09-A 11-07-LG	3.6.3.5	SR		3.6-7 3.6-12
4.6.3.4	SR		07-10-LS9 11-13-LS22	3.6.3.7	SR		3.6-13 3.6-17
3.6.4.1	LCO		12-01-A	3.3.3	LCO		
3.6.4.1	APP		12-02-M	3.3.3	APP		
3.6.4.1	ACTION	a	12-03-LS15 13-05-LS23	3.3.3	ACTION		
3.6.4.1	ACTION	b	12-04-M 13-05-LS23	3.3.3	ACTION		
4.6.4.1	SR		12-05-LS16 12-06-LG	3.3.3	SR		
	SR	New	12-07-M	3.3.3	SR		
3.6.4.2	LCO			3.6.8	LCO		
3.6.4.2	APP			3.6.8	APP		
3.6.4.2	ACTION		13-05-LS23	3.6.8	ACTION	A	
3.6.4.2		New	13-01-LS17	3.6.8	ACTION	B	
3.6.4.2	ACTION		13-05-LS23	3.6.8	ACTION	C	
4.6.4.2	SR	a	13-02-LS18 13-03-LG	3.6.8.1	SR		

CROSS-REFERENCE TABLE FOR 3/4.6
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
4.6.4.2	SR	b.1	13-02-LS18 13-04-LG			Not Used	
4.6.4.2	SR	b.2	13-02-LS18 13-03-LG	3.6.8.2	SR		
4.6.4.2	SR	b.3	13-02-LS18 13-03-LG	3.6.8.3	SR		

CROSS-REFERENCE TABLE FOR 3/4.6
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.6.1.1	LCO		01-01-LG	3.6.1	LCO		
3.6.1.6	LCO		06-02-A	3.6.1	LCO		
3.6.1.1	APP			3.6.1	APP		
3.6.1.2	APP		02-01-A	3.6.1	APP		
3.6.1.6	APP			3.6.1	APP		
3.6.1.1	ACTION		01-01-LG	3.6.1	CONDITION		
3.6.1.2	ACTION		02-01-A 02-02-A 02-06-A	3.6.1	CONDITION		
3.6.1.6	ACTION		06-04-M	3.6.1	CONDITION		
4.6.1.1	SR			3.6.1.1	SR		3.6-1
4.6.1.6.1	SR		06-02-A	3.6.1.1	SR		3.6-1
				3.6.1.2	SR		Not Used.
3.6.1.3	LCO		03-01-LG	3.6.2	LCO		
3.6.1.3	APP			3.6.2	APP		
3.6.1.2	LCO			3.6.2	CONDITION	Notes	
3.6.1.3	ACTION		03-02-A 03-09-LS7 03-13-A	3.6.2	CONDITION	Notes	
3.6.1.3	ACTION	a.1, a.2	03-03-LS3 03-04-M 03-05-LS4 03-06-LS5 03-08-LS6 03-11-LS6	3.6.2	CONDITION	A, B	
3.6.1.3	ACTION	b	03-07-M	3.6.2	CONDITION	C	
3.6.1.3	ACTION	a.3, b		3.6.2	CONDITION	D	
4.6.1.1	SR	b	01-05-A	3.6.2.1	SR		3.6-1
4.6.1.3	SR	a		3.6.2.1	SR		3.6-1
4.6.1.3	SR	b	03-10-LS8	3.6.2.2	SR		3.6-2
3.6.1.7	LCO		07-01-A	3.6.3	LCO		
3.6.3	LCO		11-01-LS13 11-11-A	3.6.3	LCO		3.6-3
3.6.1.7	APP			3.6.3	APP		
3.6.3	APP			3.6.3	APP		

CROSS-REFERENCE TABLE FOR 3/4.6
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.6.1.7	LCO		07-04-R	3.6.3	CONDITION	Notes	3.6-17
3.6.3	ACTION		03-13-A 11-02-A 11-03-A	3.6.3	CONDITION	Notes	3.6-17
3.6.3	ACTION	b, c	01-03-A 11-12-A	3.6.3	CONDITION	A.1	
4.6.1.1	SR	a	01-02-A 01-03-A 01-04-LS1 01-06-LS19	3.6.3	CONDITION	A.2	3.6-11
3.6.3		New	11-04-A	3.6.3	CONDITION	B	
3.6.3		New	11-05-LS14	3.6.3	CONDITION	C	3.6-4 3.6-11
				3.6.3	CONDITION	Not Used	
3.6.1.7	ACTION		03-13-A 07-04-R 07-05-A 11-02-A 11-12-A	3.6.3	CONDITION	D	
3.6.1.7		New	01-04-LS1 07-02-LS9	3.6.3	CONDITION	D	
3.6.3	ACTION	d		3.6.3	CONDITION	E	
				3.6.3.1	SR	Not Used	3.6-17
4.6.1.7.1	SR		07-03-A	3.6.3.2	SR		3.6-17
4.6.1.1	SR	a	01-02-A 01-04-LS1 01-06-LS19	3.6.3.3	SR		3.6-5
4.6.1.1	SR	a	01-02-A 01-04-LS1 01-06-LS19	3.6.3.4	SR		3.6-5
4.6.3.3	SR		11-09-A 11-07-LG	3.6.3.5	SR		3.6-7 3.6-12
				3.6.3.6	SR	Not Used	
4.6.3.4	SR		07-10-LS9 11-13-LS22	3.6.3.7	SR		3.6-13 3.6-17
4.6.3.2	SR	a	11-07-LG 11-14-A 11-08-TR1 11-09-A	3.6.3.8	SR		
				3.6.3.9	SR	Not Used	
4.6.1.7.3	SR			3.6.3.10	SR		

CROSS-REFERENCE TABLE FOR 3/4.6
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
				3.6.3.11	SR	Not Used	
3.6.1.4	LCO			3.6.4	LCO		
3.6.1.4	APP			3.6.4	APP		
3.6.1.4	ACTION			3.6.4	CONDITION		3.6-8
4.6.1.4	SR			3.6.4.1	SR		
3.6.1.5	LCO			3.6.5	LCO		
3.6.1.5	APP			3.6.5	APP		
3.6.1.5	ACTION			3.6.5	CONDITION		
4.6.1.5	SR		05-01-LG	3.6.5.1	SR		
3.6.2.1	LCO		08-01-LG 08-04-A	3.6.6	LCO		3.6-14
3.6.2.3	LCO			3.6.6	LCO		3.6-14
3.6.2.1	APP			3.6.6	APP		
3.6.2.3	APP			3.6.6	APP		
3.6.2.1	ACTION		08-02-A 08-11-LS2	3.6.6	CONDITION	A, B	
3.6.2.3	ACTION	a	08-11-LS2	3.6.6	CONDITION	C	3.6-14
3.6.2.3	ACTION	b		3.6.6	CONDITION	D, E	3.6-14
3.6.2.3	ACTION	New	08-10-A	3.6.6	CONDITION	F	3.6-14
4.6.2.1	SR	a		3.6.6.1	SR		
4.6.2.3	SR	a.1		3.6.6.2	SR		
4.6.2.3	SR	a.2	10-03-LG	3.6.6.3	SR		
4.6.2.1	SR	b	08-08-LG	3.6.6.4	SR		
4.6.2.1	SR	c.1		3.6.6.5	SR		
4.6.2.1	SR	c.2		3.6.6.6	SR		
4.6.2.3	SR	b	09-05-TR1	3.6.6.7	SR		3.6-14
4.6.2.1	SR	d	08-06-LG	3.6.6.8	SR		
4.6.2.3	SR	a.3		3.6.6.9	SR		3.6-14
3.6.2.2	LCO	a	09-01-A	3.6.7	LCO		

CROSS-REFERENCE TABLE FOR 3/4.6
Sorted by Improved TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.6.2.2	APP			3.6.7	APP		
3.6.2.2	ACTION		09-03-A	3.6.7	CONDITION	A, B	
4.6.2.2	SR	a		3.6.7.1	SR		
4.6.2.2	SR	b.1		3.6.7.2	SR		3.6-10
4.6.2.2	SR	b.2		3.6.7.3	SR		
4.6.2.2	SR	c	09-04-A 09-05-TR1	3.6.7.4	SR		
4.6.2.2	SR	d	09-07-M	3.6.7.5	SR		
3.6.4.2	LCO			3.6.8	LCO		
3.6.4.2	APP			3.6.8	APP		
3.6.4.2	ACTION		13-05-LS23	3.6.8	CONDITION	A	
3.6.4.2		New	13-01-LS17	3.6.8	CONDITION	B	
3.6.4.2	ACTION		13-05-LS23	3.6.8	CONDITION	C	
4.6.4.2	SR	a	13-02-LS18 13-03-LG	3.6.8.1	SR		
4.6.4.2	SR	b.2	13-02-LS18 13-03-LG	3.6.8.2	SR		
4.6.4.2	SR	b.3	13-02-LS18 13-03-LG	3.6.8.3	SR		
				3.6.9	LCO	Not Used	
				3.6.10	LCO	Not Used	
				3.6.11	LCO	Not Used	
				3.6.12	LCO	Not Used	

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

Methodology for Cross-Reference Tables
(Continued)

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).

Note: The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note: . When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
3.6.1.1	3/4 6-1
3.6.1.2	3/4 6-2
3.6.1.3	3/4 6-5
3.6.1.4	3/4 6-7
3.6.1.5	3/4 6-8
3.6.1.6	3/4 6-9
3.6.1.7	3/4 6-10
3.6.2.1	3/4 6-11
3.6.2.2	3/4 6-12
3.6.2.3	3/4 6-13
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3.6.4.2	3/4 6-18

Methodology

(2 Pages)

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 CONTAINMENT INTEGRITY shall be maintained OPERABLE.

01-01-LG

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without CONTAINMENT INTEGRITY inoperable, restore CONTAINMENT INTEGRITY to OPERABLE within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

01-01-LG

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT INTEGRITY shall be demonstrated OPERABLE

01-02-A

a. At least once per 31 days by verifying that all penetrations*# not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, or check valve with flow through the valve secured except for valves that are open under administrative control as permitted by Specification 3.6.3. Isolation devices in high radiation areas may be verified by administrative means.

01-06-LS19

1-03-A

01-04-LS1

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

01-05-A

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations# shall be verified closed during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

01-06-LS19

Except when closed by manual valves that are locked, sealed, or otherwise secured and blind flanges. If locked, sealed, or otherwise secured manual valves, blind flanges, and deactivated automatic valves are closed to satisfy an ACTION (e.g. 3.6.3) the position must be verified but may be verified by administrative means.

01-06-LS19

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

02-01-A

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the measured overall integrated containment leakage rate exceeding 1.0 L_a , within 1 hour initiate action to be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the overall integrated leakage rate to less than or equal to 0.75 L_a , and the combined leakage rate for all penetrations subject to Type B and C tests to less than or equal to 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F the first unit startup following testing performed in accordance with the Containment Leakage Rate Testing Program.

02-06-A

02-02-A

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the test schedule and shall be determined in conformance with the criteria specified in the Containment Leakage Rate Testing Program.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each ~~the two~~ containment air locks shall be OPERABLE with both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION: * ** ***

a. With one or more containment air locks with one containment airlock door or the interlock mechanism inoperable:

1. Maintain at least Verify the OPERABLE air lock door closed within 1 hour and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed, and

2. Operation may then continue until performance of the next required overall air lock leakage test provided that Verify the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the one or more containment air locks inoperable for reasons other than a, initiate Action to evaluate overall containment leakage rate per LCO 3.6.1, except as the result of an inoperable air lock door, or an interlock mechanism maintain Verify at least one air lock door closed within 1 hour, restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 36 hours.

* Entry and exit is permissible to perform repairs on the affected airlock components.

** Separate condition entry is allowed for each airlock

*** Enter applicable Conditions and Required Actions of LCO 3.6.1 "Containment" when airlock leakage results in exceeding the overall containment leakage rate.

**** Entry and exit is permissible for 7 days under Administrative controls if both airlocks are inoperable.

***** Airlock doors in high radiation areas may be verified locked by Administrative means.

++ Entry and exit of containment is permissible under the control of a dedicated individual.

03-02-A

03-13-A

03-01-LG

03-09-LS7

03-03-LS3

03-08-LS6

03-04-M

03-05-LS4

03-06-LS5

03-12-A

03-07-M

03-09-LS7

03-02-A

03-13-A

03-03-LS3

03-06-LS5

03-08-LS6

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

- a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program.
- b. At least once per 6 ~~24~~ months by verifying that only one door in each air lock can be opened at a time.

03-10-LS8

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Containment internal pressure shall be maintained between -1.0 and +1.2 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The containment internal pressure shall be determined to be within the limits at least once per 12 hours.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Containment average air temperature shall not exceed 120°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours.

05-01-LG

Element and Location

- a. ~~TE 85 or TE 86, approximately 100 ft elevation between crane wall and containment wall.~~
- b. ~~TE 87 or TE 88, approximately 100 ft elevation between steam generators.~~
- c. ~~TE 89 or TE 90, approximately 140 ft elevation near equipment hatch or stairs at 270°, respectively, and~~
- d. ~~TE 91 or TE 92, approximately 184 ft elevation on top of steam generator missile barriers away from steam generators.~~

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

06-02-A

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

06-04-M

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during shutdown by a visual inspection of these surfaces. This inspection shall be performed in accordance with the Containment Leakage Rate Testing Program to verify no apparent changes in appearance or other abnormal degradation.

06-02-A

~~4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.~~

06-03-TR2

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

07-01-A

LIMITING CONDITION FOR OPERATION

3.6.1.7 One purge supply line and/or one purge exhaust line of the Containment Purge System may be open or the vacuum/pressure relief line may be open. ~~The vacuum/pressure relief line may be open provided the vacuum/pressure relief isolation valves are blocked to prevent opening beyond 50° (90° is fully open). Operation with any two of these three lines open is permitted. Operation with the purge supply and/or exhaust isolation valves open or with the vacuum/pressure relief isolation valves open up to 50° shall be limited to less than or equal to 200 hours during a calendar year.~~

07-04-R

APPLICABILITY: MODES 1, 2, 3, and 4.

11-02-A

ACTION: ~~***~~

03-13-A

~~With a containment purge supply and/or exhaust isolation valve open or the vacuum/pressure relief isolation valves open up to 50° for more than 200 hours during a calendar year or the Containment Purge System open and the vacuum/pressure relief lines open, or with the vacuum/pressure relief isolation valves open beyond 50° With two containment purge supply or exhaust valves or two vacuum/pressure relief valves on the same penetration inoperable for reasons other than leakage, close the open isolation valve(s) or isolate the penetration(s) flowpath(s) within 1 hour; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

07-04-R

07-05-A

11-12-A

~~(new) One or more penetration flow paths with one or more containment purge or vacuum/pressure relief valves not within purge valve leakage limits. Within 24 hours isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. Verify the affected penetration flow path is isolated once per 31 days for isolation devices outside containment and prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment* and perform Surveillance 4.6.3.4 for the resilient seal purge valves closed to comply with this Required Action E-1D 1, once per 92 days.~~

07-02-LS9

01-04-LS1

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The position of the containment purge supply and exhaust isolation valves and the vacuum/pressure relief isolation valves shall be determined closed at least once per 31 days except for one valve in a penetration flow path while in action 3.6.1.7 for excessive leakage.

07-03-A

4.6.1.7.2 The cumulative time that the purge supply and/or exhaust isolation valves or the vacuum/pressure relief isolation valves have been open during a calendar year shall be determined at least once per 7 days.

07-04-R

4.6.1.7.3 The 12-inch vacuum/pressure relief isolation valves shall be verified to be blocked to prevent opening beyond 50° at least once per 18 months.

ED

* Isolation devices in high radiation areas may be verified by use of administrative means.

01-04-LS1

** Separate Condition entry is allowed for each penetration flow path.

11-02-A

*** Enter applicable Conditions and Required Actions of the Containment EGO when leakage results in exceeding the overall containment leakage rate.

03-13-A

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

08-04-A

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

8-01-LG

3.6.2.1 Two Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring spray function to a RHR System taking suction from the containment sump.

APPLICABILITY: MODES 1, 2, 3 and 4.

08-11-LS2

ACTION: *

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or ~~and~~ be in COLD SHUTDOWN within the following ~~30~~ 78 hours.

08-02-A

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

- a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;
- b. By verifying that ~~on recirculation flow, each pump's developed head at the flow test point is a differential pressure of greater than or equal to 205 psid the required developed head when tested pursuant to Specification 4.0.5 the Inservice Test Program;~~ 08-08-LG
- c. At least once per 18 months by:
 - 1) Verifying that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal, and
 - 2) Verifying that each spray pump starts automatically on an actual or simulated actuation signal.
- d. At least once per 10 years by ~~performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.~~ 08-06-LG

* ~~Additionally, a completion time of 10 days from discovery of failure to meet the conditions of 3.6.2.1 and 3.6.2.3.~~ 08-11-LS2

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

- a. A spray additive tank with a contained volume of between ~~2025 and 4000 gallons~~ 46.2% and 91.9% of between 30 and 32% by weight NaOH solution, and 09-01-A
- b. ~~Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.~~ 09-02-LG

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours. 09-03-A

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

- a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;
- b. At least once per 6 months by:
 - 1) Verifying the contained solution volume in the tank, and
 - 2) Verifying the concentration of the NaOH solution by chemical analysis.
- c. At least once per 18 months by verifying that each automatic valve in the flow path ~~that is not locked, sealed, or otherwise secured in position~~ actuates to its correct position on a ~~an actual or simulated~~ Containment Spray actuation test signal; and 09-04-A
09-05-TR1
- d. At least once per 5 years by verifying both spray additive and RWST full flow from the test valve 8993 ~~through each solution flow path in~~ the Spray Additive System. 09-07-M

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

08-04-A

LIMITING CONDITION FOR OPERATION

3.6.2.3 The Containment Cooling System shall be OPERABLE with either:

- a. At least four containment fan cooler units (CFCUs), or
- b. At least three CFCUs, each of the three supplied from a different vital bus.

APPLICABILITY: MODES 1, 2, 3, and 4.

08-11-LS2

ACTION: *

- a. With the requirements of the above specification not satisfied, but at least two CFCUs OPERABLE and both Containment Spray Systems OPERABLE, restore the Containment Cooling System to OPERABLE status within 7 days, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With the requirements of the above specification not satisfied and one Containment Spray System inoperable, but at least two CFCUs OPERABLE, restore the inoperable Containment Spray System to OPERABLE status within 72 hours otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the Containment Cooling System to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~(new) With one Containment Spray System and one or less CFCUs operable or two Containment Spray Systems inoperable enter LCO 3.0.3.~~

08-10-A

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment fan cooler unit shall be demonstrated OPERABLE:

- a. At least once per 31 days by:
 - 1) Starting each containment fan cooler unit and verifying that each containment fan cooler unit operates for at least 15 minutes.

~~* Additionally, a completion time of 10 days from discovery of failure to meet the conditions of 3.6.2.1 and 3.6.2.3.~~

08-11-LS2

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- 2) Verifying a cooling water flow rate of greater than or equal to 1650* gpm to each cooler, and 10-03-LG
- 3) Verifying that each containment fan cooler unit starts on low speed.
- b. At least once per 18 months by verifying that each containment fan cooler unit starts automatically on a Safety Injection test an actual or simulated actuation signal. 09-05-TR1

~~* The CFCU cooling water flow rate requirement of IS 4.6.2.3a.2) may not be met during Section XI testing and in Mode 4 during residual heat removal heat exchanger operation.~~

10-03-LG

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE.* #

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION: **; ***; ****

With one or more of the penetration flow paths with one isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

11-12-A

a. ~~Restore the inoperable valve(s) to OPERABLE status within 4 hours, or~~

11-16-A

b. Isolate each affected penetration flow path within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

11-12-A

c. Isolate each affected penetration flow path within 4 hours by use of at least one closed manual valve or blind flange or check valve with flow secured; or

11-12-A

01-03-A

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Not applicable to Main Steam Safety Valves (MSSVs), Main Steam Isolation Valves (MSIVs), Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs) and Associated Bypass Valves, and Atmospheric Dump Valves (ADVs)

11-11-A

NOTE 1 * ~~Locked or sealed closed valves~~ Penetration flow paths may be opened on an intermittent basis under administrative control.

11-01-LS13

(new) With one or more penetration flow paths with two containment isolation valves inoperable, isolate the affected penetration flow path within 1 hour by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

11-04-A

(new) With one or more penetration flow paths of the type configured with only one containment isolation valve and a closed system, with one containment isolation valve inoperable, isolate the affected penetration flow path within 72 hours by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

11-05-LS14

SURVEILLANCE REQUIREMENTS

4.6.3.1 ~~Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.~~

11-06-TR3

11-09-A

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each automatic containment isolation valve that is not locked, sealed or otherwise secured in position shall be demonstrated OPERABLE at least once per 18 months by:

11-14-A

11-07-LG

- a. Verifying that on a Phase "A" Isolation test an actual or simulated signal, each Phase "A" isolation valve actuates to its isolation position.
- b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and
- c. Verifying that on a Containment Ventilation Isolation test signal, each containment ventilation isolation valve actuates to its isolation position.

11-08-TR1

11-07-LG

11-07-LG

Note 2: ** Separate Condition entry is allowed for each penetration flow path.

11-02-A

*** Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.

11-03-A

**** Enter applicable Conditions and Required Actions of Specification 3.6.1.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

03-13-A

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.3 The isolation time of each testable power operated or automatic containment isolation valve that is not locked, sealed, or otherwise secured shall be determined to be within its limit when tested pursuant to Specification 4.0.5 the Inservice Testing Program.

11-07-LG

11-09-A

4.6.3.4 Each containment ventilation isolation valve, except the air sample supply and return valves, shall be demonstrated OPERABLE every 184 days and within 24 hours 92 days after each closing of opening the valve, except when the valve is being used for multiple cycling, then at least once per 72 hours, by verifying leakage rates in accordance with the Containment Leakage Rate Testing Program. This surveillance is not required when the penetration flow path is isolated by a leak tested blank flange.

11-13-LS22

07-10-LS9

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

12-01-A

HYDROGEN ANALYZERS/MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers/monitors shall be OPERABLE.

APPLICABILITY: MODES 1, and 2 and 3.

12-02-M

ACTION: ECO 3.0.4 is not applicable

13-05-LS23

- a. With one hydrogen analyzer/monitor inoperable, restore the inoperable analyzer/monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours submit a Special Report in accordance with 10 CFR 50.4 within the following 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability and the plans and schedule for restoring the hydrogen analyzer/monitor to OPERABLE status.
- b. With both hydrogen analyzer/monitors inoperable, restore at least one analyzer/monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and be in HOT SHUTDOWN within the next 6 hours.

12-03-LS15

12-04-M

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer/monitor shall be demonstrated OPERABLE at least once per 92 days 18 months by performing a CHANNEL CALIBRATION using a zero and span gas.

12-05-LS16

12-06-LG

(new) Perform CHANNEL CHECK at least once per 31 days to verify hydrogen analyzer/monitor OPERABLE.

12-07-M

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent Hydrogen Recombiner Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Recombiner System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

13-05-LS23

~~(new) With two hydrogen recombers inoperable, verify within 1 hour and once per 12 hours thereafter, by administrative means, that the hydrogen control function is maintained, and restore one hydrogen recombiner to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours.~~

13-01-LS17

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

- a. ~~At least once each refueling interval 18 months by verifying, during performing a Recombiner System functional test, that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 kW; and~~
 - 13-02-LS18
 - 13-03-LG
- b. ~~At least once each refueling interval 18 month by:~~
 - 1) ~~Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,~~
 - 13-02-LS18
 - 13-04-LG
 - 2) ~~Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and~~
 - 13-03-LG
 - 3) ~~Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.~~
 - 13-03-LG

~~* 3.0.4 is not applicable.~~

13-05-LS23

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. **Deletions** - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. **Additions** - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. **Modifications** - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. **Administrative** - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions** - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions** - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications** - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative** - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

Methodology For Mark-Up of Current TS (Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers	(1 Page)
Description of Changes	(14 Pages)

TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS

SECTION 3/4.6

Technical Specification Title	CHG NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Containment Integrity	1	3.6.1.1	3.6.1.1	3.6.1.1	3.6.1.1
Containment Leakage	2	N/A	N/A	3.6.1.2	3.6.1.2
Containment Air Locks	3	3.6.1.3	3.6.1.3	3.6.1.3	3.6.1.3
Internal Pressure	4	3.6.1.4	3.6.1.4.1	3.6.1.4.1	3.6.1.4
Air Temperature	5	3.6.1.5	3.6.1.4.2	3.6.1.4.2	3.6.1.5
Containment Structural Integrity	6	N/A	N/A	3.6.1.6	3.6.1.6
Containment Ventilation System	7	3.6.1.7	3.6.1.7	3.6.1.7	3.6.1.7
Containment Spray System	8	3.6.2.1	3.6.2.1	3.6.2.1	3.6.2.1
Spray Additive System	9	N/A	3.6.2.2	3.6.2.2	3.6.2.2
Recirculation Fluid pH Control (RFPC) System	9	3.6.2.2	N/A	N/A	N/A
Containment Cooling System	10	3.6.2.3	3.6.2.3	N/A	3.6.2.3
Containment Isolation Valves	11	3.6.3	3.6.3	3.6.3	3.6.3
Hydrogen Analyzers/ Monitors	12	N/A	N/A	3.6.4.1	3.6.4.1
Electric Hydrogen Recombiners	13	N/A	N/A	3.6.4.2	3.6.4.2
Hydrogen Control System	13	3.6.4.2	3.6.4.2	N/A	N/A

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	LG	CONTAINMENT INTEGRITY is no longer a defined term in NUREG-1431. The requirements for containment OPERABILITY, including the requirements previously found in the CONTAINMENT INTEGRITY definition, are discussed in the expanded Bases of the containment limiting condition for operation (LCO). This change is consistent with NUREG-1431.
01-02	A	Consistent with NUREG-1431, this requirement to verify the penetration flow path is isolated is now addressed by improved Technical Specification (ITS) 3.6.3, Containment Isolation Valves, Condition A, C, and [D] and Surveillance Requirements (SR) 3.6.3.3 and SR 3.6.3.4.
01-03	A	The ACTION statements are revised to incorporate the NUREG-1431 alternative isolation method of a "check valve with the flow through the valve secured." This isolation method is provided in NUREG-1431 and is considered an acceptable variation of a de-activated automatic valve.
01-04	LS1	A Note is added to valve and blind flange SRs consistent with NUREG-1431. The Note allows verification of valves, flanges, and isolation devices located in high radiation areas to be verified by use of administrative means. This change is less restrictive in that the CTS SR 4.6.1.1 has an exception to valves, blind flanges, and deactivated automatic valves which are located inside containment and are locked, sealed, or otherwise secured in the closed position. These valves shall be verified closed during each COLD SHUTDOWN. However, under CTS, if an area outside of containment becomes a high radiation area, we would still be required to enter the area to verify the closed positions. The ITS would allow verification of all areas that are high radiation areas or become high radiation areas to be verified by administrative means once they have been verified to be in the proper position. This is considered acceptable, since access to these areas is restricted for ALARA reasons. Therefore, the probability of misalignment of these devices, once they have been initially verified in the proper position, is small.
01-05	A	Consistent with NUREG-1431, this requirement is addressed by SR 3.6.2.1 in ITS 3.6.2, Containment Air Locks Required Actions.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-06	LS19	<p>The CTS requires all penetrations not capable of being closed by an OPERABLE containment automatic isolation valve (and required to be closed for accident conditions) be verified closed on a 31 day frequency, except for valves, blind flanges, and deactivated automatic valves which are located inside containment and locked closed, sealed, or otherwise secured in a closed position. These excepted penetrations were to be verified closed during each cold shutdown but not more often than once per 92 days. Consistent with Traveler TSTF-45, Rev. 1 only containment isolation valves that are not locked, sealed, or otherwise secured are required to be verified closed at the once per COLD SHUTDOWN frequency. Penetrations (inside or outside containment) which are isolated by manual valves and blind flanges that are locked, sealed or otherwise secured are not required to be verified closed, since they are verified to be in the correct position prior to locking/securing.</p> <p>CTS Surveillance 4.6.1.1 is incorporated into ITS SR 3.6.3.3 and SR 3.6.3.4 and into the redundant requirements of ITS 3.6.3, Required Action A.2, C.2, and [D.2]. As discussed above, the ITS surveillances were modified consistent with TSTF-45. The redundant Required Actions were also modified in a subsequent Traveler (WOG-91). The TSTF-45 modifications to the ITS surveillances and the WOG-91 modifications to Required Actions, while similar, differed in that the Required Actions still require some verification of position of valves that are locked closed, sealed or otherwise secured, although this verification may be performed via administrative means. The surveillances require no additional verification.</p>
02-01	A	Consistent with NUREG-1431, the Containment Leakage LCO is now included in ITS 3.6.1, Containment LCO.
02-02	A	The wording "prior to increasing the Reactor Coolant System temperature above 200°F" is replaced by the equivalent requirement of "prior to the first unit startup following testing performed in accordance with the Containment Leakage Rate Testing Program." The fact that the Applicability of the new Containment LCO is MODE 1-4, and that SR 3.0.4 requires surveillances to be performed before entering the MODE of Applicability, ensures that the required leakage rate testing is performed and that the as-left test acceptance criteria are met before entry into MODE 4. This change is consistent with NUREG-1431. This requirement is now included in ITS 5.5.16, the Containment Leak Rate Test Program.
02-03	A	This change is not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
02-04	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-05	LG	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
02-06	A	Consistent with industry Traveler TSTF-52, the leakage rate acceptance criteria is revised to $<0.60 L_a$ for Type B and Type C tests rather than $\leq 0.60 L_a$. There is no significant difference between the two expressions.
03-01	LG	The descriptive information in the LCO regarding OPERABILITY of the air locks is moved to the Bases 3.6.2. This change is consistent with NUREG-1431 and the approach of discussing OPERABILITY in the TS Bases.
03-02	A	This Note revises ACTIONS to permit separate Condition entry for each airlock consistent with NUREG-1431. The Note provides guidance and clarification for use of the TS and is considered administrative in nature.
03-03	LS3	Consistent with NUREG-1431, this Note allows entry and exit into containment via the air locks for up to 7 days if both air locks are inoperable and administrative controls are provided. With both air locks inoperable containment entry may be required on a periodic basis to perform TS surveillances and Required Actions, as well as other activities on equipment inside containment. This Note provides allowance for these activities to be performed.
03-04	M	The CTS ACTIONS [] are revised to be consistent with NUREG-1431, LCO 3.6.2, ACTIONS A.1. and A.2. The NUREG-1431 ACTIONS establish a one hour time limit for verifying the OPERABLE air lock door closed. The current requirement does not specify a time for verifying an air lock door closed.
03-05	LS4	The allowance to continue operation with one air lock door inoperable has been modified to remove the restriction which limits this Condition until the next required overall air lock leakage test. This restriction was removed because the air lock remains capable of performing its safety functions with the remaining OPERABLE door. Therefore, continued operations may proceed indefinitely subject to the other restrictions of the TS (continuing to meet the ACTIONS and applicable surveillances). This change is consistent with NUREG-1431.
03-06	LS5	Consistent with NUREG-1431, this Note modifies the requirement to verify an air lock door locked closed every 31 days. The Note allows the verification of locked closed air lock doors located in high radiation areas to be performed by use of administrative means. This change is less restrictive in that the CTS does not require this exception due to current design and capability to verify inner door locked from outside of the containment airlock. Under CTS, if the area outside of the airlock and containment became a high radiation area, we would still be required to enter the area to verify the closed positions. The ITS would allow verification of all areas that are high radiation areas or become high radiation areas by administrative means once they have been verified to be in the proper position. This is considered acceptable, since access to these areas is restricted for ALARA reasons. Therefore, the probability of misalignment of the air lock doors once they have been initially verified in the proper position, is small.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)

CHANGE NUMBER

NSHC

DESCRIPTION

03-07

M

Consistent with NUREG-1431, this ACTION addresses any air lock inoperability other than an inoperable air lock door or air lock interlock mechanism. This includes two air lock doors in the same air lock. A new TS requirement, that is current operating practice conducted under the Containment Leakage Rate Testing Program, is included to immediately initiate ACTION to evaluate the overall containment leakage rate per the containment OPERABILITY LCO if the air lock limit has been exceeded. In addition, this change establishes a one hour time limit to verify one air lock door is closed in the affected air lock. The CTS does not specify a time.

03-08

LS6

Consistent with the ACTIONS and modifying Notes provided for an inoperable airlock door, ACTIONS are added which allow continued operation when the personnel airlock is inoperable due to an inoperable interlock mechanism. Use of the airlock is permissible under the control of a dedicated individual since an equivalent level of assurance that only one door will be open at a time is provided. The ACTIONS and modifying Notes provided for the interlock mechanism are consistent with the ACTIONS and Notes for an inoperable airlock door. These ACTIONS ensure that an OPERABLE door is closed and locked and entry or exit is controlled by a dedicated individual. While this change provides a slightly less restrictive TS requirement, the overall impact on plant safety is negligible due to the Required Actions implemented.

03-09

LS7

Consistent with NUREG-1431, a Note is added to allow access (entry and exit) to repair airlock components in an inoperable airlock. The ACTION requirement to close and lock an airlock door remains applicable with the exception of the brief entry and exit of the airlock to perform necessary repairs. The allowance to enter and exit through the airlock door required closed by the ACTIONS is acceptable based on the low probability of an event occurring that could challenge the containment boundary during the short time the door is open for entry and exit. In addition, this allowance reduces the potential risk incurred during a plant transient that may result from a shutdown required by TS due to the inoperable airlock.

03-10

LS8

Consistent with industry Traveler TSTF-17, Rev. 1, the surveillance Frequency on containment airlock interlock mechanisms is extended from 6 months to 24 months. The 24 month Frequency is based on engineering judgement and is considered adequate given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit. The 24 month Frequency is needed to allow the surveillance to be performed under conditions that apply during a plant outage due to the potential for loss of containment OPERABILITY if the surveillance were performed with the reactor at power.

03-11

Not Used.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6

(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
03-12	A	The statement that Specification 3.0.4 does not apply is no longer needed as revised ACTIONS consistent with NUREG-1431 permit continued operation for an unlimited period of time.
03-13	A	Consistent with NUREG-1431, a Note is added to the ACTIONS to enter applicable Conditions and Required ACTIONS of the "Containment" LCO when leakage results in exceeding the overall containment leakage rate. This is current operating practice per TS requirement CTS 3.6.1.1. Therefore, this change is considered an administrative change in format.
05-01	LG	The method for calculating containment average temperature and the locations where measurements are taken are moved to the Bases. This level of detail in the TS is not consistent with NUREG-1431. The improved STS Bases is licensee controlled under the Bases Control Program in the Administrative Controls section of the improved STS.
06-01		Not Used.
06-02	A	The structural integrity requirements of containment are contained in ITS 3.6.1. The inspection requirements associated with structural integrity of the exposed accessible interior and exterior containment surfaces, are contained in Appendix J, Option B and in Regulatory Guide (RG) 1.163. The requirement to perform visual inspections is in ITS Surveillance Requirement (SR) 3.6.1.1 which refers to the containment Leakage Rate Testing Program as controlled by ITS 5.5.16.
06-03	TR2	Consistent with NUREG-1431, the reporting requirement is being deleted. 10 CFR 50.72 and 10 CFR 50.73 establish the reporting requirements.
06-04	M	The ACTION is moved to ITS 3.6.1, Condition A and B. The ITS requirements are more severe in that only 1 hour allowed outage time (AOT) is provided while the CTS provides a 24 hour AOT. The shorter AOT is acceptable because a containment which may not be able to act as a boundary as designed could have a significant adverse impact on the consequences of an accident.
07-01	A	Consistent with NUREG-1431, the LCO and SRs for containment ventilation/purge valves are now included in ITS 3.6.3 for Containment Isolation Valves.
07-02	LS9	Consistent with NUREG-1431, the Required Action for a containment ventilation/purge valve with a leakage rate which exceeds the acceptance criteria is revised to allow continued operation if the penetration flow path is isolated within 24 hours. This action is in lieu of requiring a shutdown if the valve leakage rate is not restored to an acceptable value within 24 hours. This is considered acceptable because with the associated penetrations isolated per the proposed ACTION requirement, no accident is credible as a result of the leaking valve.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
07-03	A	A change [] consistent with NUREG-1431, SR 3.6.3.1, is added to provide an allowance for one isolation valve in a penetration flow path to be open when performing the Required Actions for leakage not within limits. This is actually a consistency change which goes with the revised Required Actions (see 07-02-LS9 above). The Required Action allows continued operation with leakage not in limits and this change to the SR allows a valve to be opened to repair the excessive leakage.
07-04	R	The time limit restrictions on opening the [containment purge supply and exhaust and pressure/vacuum relief flow paths] and the requirements to periodically accumulate the time that the valves have been open would be relocated to licensee controlled documents.
07-05	A	Consistent with NUREG-1431, an ACTION is added for two valves inoperable in one penetration flow path. The change is administrative since the CTS would have relied on LCO 3.0.3 which has essentially the same requirements.
07-06	LS11	This change is not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-07	LG	This change is not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-08	M	This change is not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-09	LG	This change is not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-10	LS9	A Note is added to clarify that leakage rate testing is not required for containment purge valves with resilient seals when the penetration flow path is isolated by a leak tested blank flange. The purpose of the leak testing requirement is to ensure containment leakage integrity during an accident, and thereby limit accident consequences. Isolation of the flow path with a leak tested blind flange accomplishes this safety function and additional leak testing of the valves in the flow path is redundant and unnecessary.
07-11	LS25	This change is not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
08-01	LG	Consistent with NUREG-1431, the LCO references to suction flowpath and manual transfer of suction to containment sump have been deleted. These details are included within the OPERABILITY requirements of the containment spray system (CSS) (as required by CTS 4.6.2.1 and as further described in the Bases). There is no technical change resulting from this deletion.

**DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)**

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
08-02	A	Consistent with NUREG-1431, the ACTION statement is revised by replacing the reference to restoring the CSS to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the following 30 hours, with the requirement to be in COLD SHUTDOWN within 78 hours. The time allowed to be in cold shutdown has not changed. The requirements of the action statement have also not changed, since as discussed in the Bases, the extended interval to reach COLD SHUTDOWN allows 48 hours for restoration of the system OPERABILITY and an additional 36 hours to achieve COLD SHUTDOWN.
08-03	TR1	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	The LCOs for containment spray system and containment coolers are combined into one LCO per NUREG-1431.
08-05	LS12	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-06	LG	The details on flow testing for nozzle obstructions and specific actuation signals that apply for automatic actuations are moved to the Bases. This is acceptable as the requirement to test remains in the Technical Specification and this level of detail is not contained in NUREG-1431.
08-07		Not Used.
08-08	LG	The specific pump discharge pressure value would be moved to the Inservice Testing Program.
08-09	LG	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-10	A	ITS Condition F. specifies two containment spray trains or any combination of three or more trains inoperable to enter 3.0.3. Even though this condition is not specified in the CTS, 3.0.3 would be entered.
08-11	LS2	A "from discovery of failure to meet the LCO provision" has been added to the Completion Time for one train of containment spray/cooling systems inoperable. This change is considered less restrictive in that the 10 days allowed in the ITS not to meet the LCO is greater than the CTS would allow.
09-01	A	The DCPP units for the spray additive tank volume limits are changed from gallons to percent.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
09-02	LG	The descriptive information in LCO 3.6.2.2 regarding OPERABILITY of the spray additive system is contained within the definition of OPERABILITY as described in the ITS 3.6.7 Bases. This is consistent with NUREG-1431 and is acceptable because while the descriptive detail has been moved to the Bases, the basic requirement is retained in the LCO.
09-03	A	Consistent with NUREG-1431, the ACTION statement is revised by deleting the reference to restoring the spray additive system to OPERABILITY within 48 hours or be in COLD SHUTDOWN within the following 30 hours. The revised ACTION statement contains a requirement to be in COLD SHUTDOWN within 78 hours. The time allowed to be in COLD SHUTDOWN has not changed. As discussed in the Bases, the interval to reach COLD SHUTDOWN allows 48 hours for restoration of the system OPERABILITY and an additional 36 hours to achieve COLD SHUTDOWN.
09-04	A	Consistent with NUREG-1431, adds the phrase "that is not locked, sealed, or otherwise secured in position with regard to which valves require actuation testing. This change is merely a clarification. Valves that are secured in place, are secured in the position required to meet their safety function. The actuation testing ensures that valves can move to the position that meets their safety function. If the valves are secured in the position that meets their safety function, no testing is necessary.
09-05	TR1	The specific actuation signal (a safety injection test signal) for the surveillance was replaced with a generic words that allow credit for an actual or simulated actuation. Identification of the signal is moved to the Bases.
09-06	LG	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
09-07	M	Consistent with NUREG-1431, the surveillance is modified to require demonstration of flow through each solution flow path. This assures that all spray additive flow paths are clear.
10-01	LG	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-02	LG	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-03	LG	The DCPP specific Note cautioning that containment fan cooling unit (CFCU) flow rate may not be achieved during Section XI testing and residual heat removal (RHR) operation is relocated to the Bases. This level of detail is not required in the TS.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
11-01	LS13	Consistent with NUREG-1431, LCO 3.6.3, a Note is added to the ACTIONS that apply to all containment isolation valves. The first Note allows containment isolation valves that are required to be closed [] to be opened under administrative controls. This is acceptable based on the administrative controls consisting of a dedicated operator at the valve in continuous communication with the control room. This control provides protection equivalent to the automatic isolation system. [] Opening on an intermittent basis under administrative controls is allowed for certain valves by references made from CTS 1.8 and 4.6.1.1 of the CTS.
11-02	A	Consistent with NUREG-1431 LCO 3.6.3, a Note is added to the ACTIONS to allow separate Condition entry for each penetration flow path. The Note provides guidance and clarification for the use of TS and is consider administrative in nature.
11-03	A	This Note is added to the ACTIONS to enter applicable Conditions and Required ACTIONS for systems made inoperable by containment isolation valves.
11-04	A	NUREG-1431 adds a new Condition to the current Containment Isolation Valve TS to cover the case where two containment isolation valves in a penetration flow path are inoperable. The CTS addresses only the condition of one valve inoperable in a penetration flow path. If two valves were inoperable on the same penetration, LCO 3.0.3 would be entered. Consistent with NUREG-1431, a completion time of 1 hour is provided to isolate the penetration flow path. This is the same amount of time allowed by LCO 3.0.3 before a power reduction ACTION is specified and is administratively similar to the existing requirements.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6

(Continued)

CHANGE NUMBER

NSHC

DESCRIPTION

11-05

LS14

A new Condition is added to the current Containment Isolation Valve TS to cover the case where one containment isolation valve is inoperable in a penetration flow path of the type configured with only one containment isolation valve and a closed system. General Design Criteria (GDC) 57 allows the use of a closed system in combination with a containment isolation valve to provide the two containment barriers against the release of radioactive material following an accident. Traveler TSTF-30, Rev. 1, provides the basis for extending the Completion Time for this Condition to allow 72 hours to isolate an inoperable isolation valve associated with a closed system. The CTS, LCO 3.6.3, does not allow the use of a closed system to isolate a failed containment isolation valve even though the closed system is subjected to a Type A containment leakage test, is missile protected, and is seismic Category I piping. Also, a closed system typically has flow through it during normal operation such that any loss of integrity could be observed through leakage detection systems within containment and system walkdowns for closed systems outside containment. As such, the use of a closed system is equivalent to isolating a failed containment isolation valve by use of a single valve as specified in NUREG-1431, Required ACTION A.1. The 72 hours provides the necessary time to perform repairs on a failed containment isolation valve when relying on an intact closed system. A Completion Time of 72 hours is considered appropriate given the reliability of the closed system and that 72 hours is typically provided for losing one train of redundancy throughout the NUREG. If the closed system and associated containment isolation valve were both inoperable, the plant would be in LCO 3.0.3 since there is no specific Condition specified.

11-06

TR3

Consistent with NUREG-1431, the CTS SR to demonstrate the OPERABILITY of each containment isolation valve by performance of a cycling and isolation time test prior to returning the valve to service after maintenance, repair, or replacement work on the valve or its associated actuator, control or power circuit has been deleted. Any time repairs, maintenance or modifications have affected the OPERABILITY of a system or component, post-maintenance testing is required to demonstrate operability of the system or component. Particular SRs needed to demonstrate OPERABILITY of the system must be evaluated for each maintenance or modification performed. Explicit post-maintenance and modification TS SRs have therefore been deleted from the ITS because these requirements are inherent in the LCO OPERABILITY requirements.

11-07

LG

Consistent with the NUREG-1431 level of detail, the descriptive material regarding the required containment isolation valve actuation signals in the CTS surveillance requirement is moved to the revised expanded Bases. This is acceptable as the requirement to verify actuation of the valves is retained in the TSs while the identification of the applicable actuation signal is moved to the Bases.

11-08

TR1

The actuation surveillance is revised consistent with NUREG-1431 to clarify that an actual signal as well as a test signal may be used to verify actuation. The actuation signal is moved to the Bases.

**DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)**

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
11-09	A	Consistent with NUREG-1431 and industry Traveler TSTF-46, Rev. 1, the isolation time surveillance is revised to delete the reference to verifying "each power operated" containment isolation valve and only require verification of each "automatic isolation valve." Containment isolation valves which are power operated but do not receive a containment isolation signal (i.e. can be remotely operated), do not have an isolation time assumed in the accident analysis since they require operator action. Therefore, deleting the reference to power operated isolation valve time testing is a clarification that reduces the potential for misinterpreting the requirements of this SR while maintaining the assumptions of the accident analysis.
11-10	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
11-11	A	A Note is added to the containment isolation specification that the LCO is not applicable to main steam safety valves (MSSVs), main steam isolation valves (MSIVs), feedwater isolation valves (FIVs), [] and atmospheric dump valve (ADVs). License Amendment (LA) 73/72 (LAR 91-08, 12/26/91) removed the listing of containment isolation valves (Table 3.6-1) and authorized revision of the list under the control of the Administrative section of the TS (e.g., under 10 CFR 50.59). These valves are currently not considered to have a containment isolation function. This note is consistent with current licensing bases.
11-12	A	The phrase "flow path" is added for clarification and constancy with NUREG-1431. This specification is based on GDC 55, 56, and 57 which address the proper isolation for each "line" that penetrates containment. Licensees have always been required to assure that proper protection is provided for each line or flow path that passes through containment even if multiple flow paths share the same penetration. In this specification, the term "penetration" has always meant each flow path that penetrates containment. Adding the words "flow path" to the specification clarifies this meaning.
11-13	LS22	This change revises the DCPP containment ventilation isolation valve leak rate surveillance frequency from 30 months to every 184 days and from 24 hours to 92 days after opening a valve. This change is consistent with NUREG-1431 and NRC resolution of Multi-Plant Action No. B-20, "Containment Leakage Due to Seal Deterioration." These valves have a good service record and have consistently met leakage rate requirements. The revised 92 day frequency still reflects conservative margin to compensate for degradation of the resilient seats in these valves.

**DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)**

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
11-14	A	Consistent with NUREG-1431, the phrase "that is not locked, sealed, or otherwise secured in position" is added for clarification in regard to which valves require isolation time testing. Valves that are secured in place, are secured in the position required to meet their safety function. The isolation time testing ensures that valves can respond to the position that meets their safety function in the time assumed in the safety analysis. If the valves are secured in the position that meets their safety function, no testing is necessary.
11-15	A	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
11-16	A	Even though it is not specified in ITS 3.6.3 Required Actions, the ACTION to restore the inoperable valve stated in CTS 3.6.3.a is understood as always the primary objective and a continuous option to be performed during any Completion Time.
11-17		Not Used.
11-18	LG	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-01	A	Consistent with NUREG-1431, the hydrogen monitoring specification is moved to improved STS Section 3.3.3 concerning Post Accident Monitoring (PAM) Instrumentation.
12-02	M	Consistent with the MODE of Applicability for PAM instrumentation in NUREG-1431, the MODE of Applicability for the hydrogen monitors is extended to MODE 3.
12-03	LS15	Consistent with NUREG-1431 the ACTION is revised to require a special report be submitted within 14 days in lieu of being in HOT STANDBY within 6 hours, if one train of hydrogen monitoring cannot be restored to operable within 30 days. This is acceptable because the report is required to identify alternative methods for monitoring and plans and schedule for restoring the instrumentation before a loss of functional capability occurs. (See ITS LCO 3.3.3 and 5.6.8)
12-04	M	Adds the requirement to be in HOT SHUTDOWN within 12 hours if both trains of hydrogen monitoring are inoperable and one train was not restored within 72 hours. This is a more restrictive requirement than is currently applied and is being done to be consistent with the PAM requirements in NUREG-1431.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
12-05	LS16	Revises the Frequency of the hydrogen monitor surveillance to perform CHANNEL CALIBRATION from 92 days on a staggered test basis to once per 18 months consistent with NUREG-1431. The hydrogen monitors are part of the PAM instrumentation and their primary function is to detect high hydrogen concentration conditions that may occur during accident situations. This change is acceptable because the primary means of reducing hydrogen concentration during accidents is via the independent hydrogen recombiners [and hydrogen purge systems]. Failure of the monitors would not affect the capabilities of [these systems]. Further changing the CHANNEL CALIBRATION surveillance interval from 92 days (on a staggered test basis) to every [18 months] is not expected to effect the reliability or performance of the hydrogen monitors based on industry operating experience.
12-06	LG	The details provided for performing the CHANNEL CALIBRATION are moved out of the SR. This information is procedural in nature and is not consistent with the level of detail in NUREG-1431. The information is moved to the Bases for ITS SR 3.3.3.2.
12-07	M	A new SR is added for DCPD requiring a CHANNEL CHECK every 31 days (if energized) for the hydrogen analyzer/ monitors. This change is consistent with NUREG-1431.
13-01	LS17	A new Condition has been added to this specification. This Condition describes the Required Action for two hydrogen recombiners inoperable. Whereas in the current specification LCO 3.0.3 applied, this change allows up to 7 days to restore one hydrogen recombiner to OPERABLE status, based on the availability of the containment hydrogen purge system to provide the required safety function. In order to use this ACTION time, the Required Actions require that the hydrogen control function be verified available within 1 hour and once every 12 hours thereafter. This administrative verification will assure that the hydrogen purge system is capable of performing the safety function if an event occurs. Also, the Bases for operation of the recombiners indicates that if a design basis event occurs, 8 days or more would elapse before the containment atmosphere approached the lower flammability limit for hydrogen. Therefore, it is reasonable to assume that the inoperability of two hydrogen recombiners will not significantly jeopardize the capability of the facility to respond to a design basis event. This change is consistent with NUREG-1431.
13-02	LS18	The current SR to perform a hydrogen recombiner functional test every 6 months is revised to every 18 months consistent with NUREG-1431. This change is considered acceptable due to the redundancy and proven high reliability of the system. Hydrogen recombiner operating experience has shown that functional test failures are rare. In addition, the fully redundant and independent hydrogen purge system provides an alternate, and equally effective, method of controlling hydrogen. The proposed change is in accordance with NUREG-1366, "Improvement to Technical Specification Requirements" and NUREG-1431.
13-03	LG	Descriptive information regarding the current hydrogen recombiner surveillances is moved into the Bases. The proposed changes to the surveillances are consistent with the wording and detail present in the NUREG-1431 surveillance requirements.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)

CHANGE NUMBER

NSHC

DESCRIPTION

13-04

LG

The SR to perform a CHANNEL CALIBRATION on all the hydrogen recombiner instrumentation is moved to a Licensee controlled document in accordance with NUREG-1431. These calibrations and any necessary compensatory measures (i.e., substitute test instrumentation) will be controlled administratively by the plant preventive maintenance and operational procedures. This change is acceptable based on the system redundancy, available alternate means of controlling hydrogen, the fact the recombiners are controlled manually, and the instrumentation does not provide essential control or interlock function. In addition, the functional test required by the TS every 18 months verifies the operation of the hydrogen recombiner instrumentation. This change is consistent with NUREG-1431.

13-05

LS23

Added statement that LCO 3.0.4 is not applicable to ACTION. This allowance is based upon the pressure of another 100% hydrogen recombiner, the hydrogen purge system and the time available for operator action after a LOCA.

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(12 pages)

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.6

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 LG	CONTAINMENT INTEGRITY is no longer a defined term in NUREG-1431. The requirements for containment OPERABILITY, including the requirements previously found in the CONTAINMENT INTEGRITY definition, are discussed in the expanded Bases of the containment LCO.	Yes	Yes	Yes	Yes
01-02 A	Consistent with NUREG-1431, this SR to verify the penetration flow path is isolated is addressed by ITS 3.6.3, Containment Isolation Valves.	Yes	Yes	Yes	Yes
01-03 A	An equal alternative isolation method of a "check valve with the flow through the valve secured" is added to the ACTION statements.	Yes	Yes	Yes	Yes
01-04 LS1	A note is added allowing valves, flanges, and isolation devices located in high radiation areas to be verified by use of administrative means.	Yes	Yes	Yes	Yes
01-05 A	This requirement is addressed by 3.6.2, Containment Air Locks Required Actions.	Yes	Yes	Yes	Yes
01-06 LS19	Only containment isolation valves that are not locked, sealed, or otherwise secured are required to be verified closed.	Yes	Yes	Yes	Yes
02-01 A	The Containment Leakage LCO is now addressed by ITS 3.6.1, Containment.	Yes	Yes	Yes	Yes
02-02 A	The wording "prior to increasing the Reactor Coolant System temperature above 200°F" is replaced by the equivalent requirement of "prior to the first unit startup following testing performed in accordance with the Containment Leakage Rate Testing Program."	Yes	Yes	No, 3.6.1.2 not in CTS.	No, 3.6.1.2 not in CTS.
02-03 A	CPSES testing requirements for containment air locks are now provided in ITS 3.6.2 for Containment Air Locks.	No	Yes	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.6

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-04 A	LCO and SR for containment ventilation/purge valves are now included in ITS 3.6.3 for Containment Isolation Valves.	No, CTS does not contain this requirement.	Yes	No, 3.6.1.2 not in CTS.	No, 3.6.1.2 not in CTS.
02-05 LG	The descriptive material contained in the CPSES CTS identifying those valves which are or may be tested with water is moved to the 3.6.1 Bases.	No	Yes	No	No
02-06 A	Consistent with industry Traveler TSTF-52, the leakage rate acceptance criteria is revised to <0.60 La for Type B and Type C tests.	Yes	Yes	No, 3.6.1.2 not in CTS.	No, 3.6.1.2 not in CTS.
03-01 LG	The descriptive information in the LCO regarding OPERABILITY of the air locks is moved to the 3.6.2 Bases.	Yes	Yes	Yes	Yes
03-02 A	This Note revises ACTIONS to permit separate Condition entry for each airlock consistent with NUREG-1431. The Note provides guidance and clarification for use of the TS and is considered administrative in nature.	Yes	Yes	Yes	Yes
03-03 LS3	This Note allows entry and exit into containment via the air locks for up to 7 days if both airlocks are inoperable and administrative controls are provided.	Yes	Yes	Yes	Yes
03-04 M	This action is revised to establish a one hour time limit for verifying an inoperable airlock door closed.	Yes	Yes	Yes	Yes
03-05 LS4	The allowance to continue operation with one airlock door inoperable is modified to remove the restriction which limits this Condition until the next required overall airlock leakage test.	Yes	Yes	Yes	Yes
03-06 LS5	This note modifies the requirement to verify an airlock door locked closed every 31 days. The note allows the verification of locked closed airlock doors located in high radiation areas to be performed by use of administrative means.	Yes	Yes	Yes	Yes

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-07 M	This ACTION addresses Conditions other than a single inoperable airlock door or airlock interlock mechanism in an affected airlock. A new requirement is included to immediately initiate ACTION to evaluate the overall containment leakage rate per the containment OPERABILITY LCO. This change also establishes a one hour time limit to verify one airlock door closed in the affected airlock.	Yes	Yes	Yes	Yes
03-08 LS6	Consistent with the ACTIONS and modifying notes provided for an inoperable airlock door, ACTIONS are added which allow continued operation when the personnel airlock is inoperable due to an inoperable interlock mechanism.	Yes	Yes	Yes	Yes
03-09 LS7	A note is added to allow entry and exit to repair airlock components in an inoperable airlock.	Yes	Yes	Yes	Yes
03-10 LS8	Consistent with industry Traveler TSTF-17, Rev. 1, the surveillance frequency on containment airlock interlock mechanisms is extended from 6 months to 24 months.	Yes	Yes	Yes	Yes
03-11	Not Used.	N/A	N/A	N/A	N/A
03-12 A	The statement that Specification 3.0.4 does not apply is not needed as revised ACTIONS permit continued operation for an unlimited period of time.	Yes	Yes	Yes	Yes
03-13 A	A Note is added to the ACTIONS to enter applicable Conditions and Required Actions of the "Containment" LCO when leakage results in exceeding the overall containment leakage rate.	Yes	Yes	Yes	Yes
05-01 LG	The method for calculating containment average temperature and the locations where measurements are taken are moved to the Bases.	Yes	Yes	Yes	Yes

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
06-01	Not used	N/A	N/A	N/A	N/A
06-02 A	The inspection requirements associated with structural integrity of the exposed accessible interior and exterior containment surfaces, are contained in Appendix J, Option B and in RG 1.163.	Yes	Yes	No, 3.6.1.6 not in CTS.	No, 3.6.1.6 not in CTS.
06-03 TR2	Reporting requirement for containment structural integrity are deleted.	Yes	Yes	No, 3.6.1.6 not in CTS.	No, 3.6.1.6 not in CTS.
06-04 M	AOT for containment structural integrity not established decreased from 24 hours to 1 hour.	Yes	Yes	No, 3.6.1.6 not in CTS.	No, 3.6.1.6 not in CTS.
07-01 A	The LCO and SRs for containment ventilation/purge valves are now included in ITS 3.6.3 for Containment Isolation Valves.	Yes	Yes	Yes	Yes
07-02 LS9	The Required Actions for a containment ventilation/purge valve with a leakage rate which exceeds the acceptance criteria is revised to be stated on a per penetration flow path bases.	Yes	Yes	Yes	Yes
07-03 A	Clarification is added to allow one isolation valve in a penetration flow path to be opened for repairs when performing the Required Actions for leakage rate not within limits.	Yes	Yes	Yes	Yes
07-04 R	The time limit restrictions on opening the [pressure/vacuum relief] and the requirements to periodically accumulate the time that the valves have been open would be relocated to licensee controlled documents.	Yes, relocated to an ECG.	No, CPSES does not have restrictions on these valves.	Yes, relocated to USAR Chapter 16	Yes, relocated to FSAR Chapter 16
07-05 A	An ACTION is added for two valves inoperable in one penetration flow path.	Yes	Yes	Yes	Yes

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
07-06 LS11	The leakage rate testing frequency for containment isolation valves with resilient seals is revised to 184 days and testing on a staggered test basis is no longer required. A new requirement is also added to perform a leakage test within 92 days of opening the valves.	No, see 11-13-LS22.	Yes	Yes	Yes
07-07 LG	The leakage rate test acceptance criteria for containment isolation valves with resilient seals is moved to the Bases.	No, criteria already moved from CTS.	Yes	Yes	Yes
07-08 M	Adds new requirement to perform a 31 day surveillance to verify closure of the 18-inch the mini-purge valve.	No, CTS already contains this requirement.	No, CPSES does not have restrictions on these valves.	Yes	Yes
07-09 LG	Details regarding the valve size and isolation requirements have been moved to the ITS Bases.	No, this detail is not in the CTS.	No, this detail is not in the CTS.	Yes	Yes
07-10 LS9	A note is added to clarify that leakage rate testing is not required for containment purge valves with resilient seals when the penetration flow path is isolated by a leak tested blank flange.	Yes	Yes	Yes	Yes
07-11 LS25	Removes the requirement to blank flange the containment shutdown purge supply and exhaust isolation valves and revises the SR for verification of closed shutdown purge valves and flanges inside containment.	No, CTS does not contain this requirement.	No, CTS does not contain this requirement.	Yes	Yes
08-01 LG	The LCO references to suction flow path and manual transfer of suction to containment sump have been deleted. These details are included within the OPERABILITY requirements of the CSS as described in the Bases.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.6

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
08-02 A	This change revises the ACTION statement by replacing the reference to restoring the CSS to operable status within 48 hours or be in COLD SHUTDOWN within the following 30 hours, with the requirement to be in COLD SHUTDOWN within 78 hours.	Yes	Yes	Yes	Yes
08-03 TR1	The actuation surveillance is revised to clarify that an actual signal as well as a test signal may be used to verify actuation. The specific actuation signals that apply for automatic actuation are moved to the Bases.	No, LA 114/112 made this part of CTS.	Yes	Yes	Yes
08-04 A	The LCOs for CSS and containment coolers are combined into one LCO per NUREG-1431.	Yes	No, CPSES CTS does not have a containment cooler specification.	Yes	Yes
08-05 LS12	This CPSES specific change revises the frequency of the surveillance to verify unobstructed spray nozzles, from every 5 years to every 10 years.	No	Yes	No	No
08-06 LG	The details on flow testing for nozzle obstructions and specific actuation signals that apply for automatic actuations are moved to the Bases.	Yes	Yes	Yes	Yes
08-07	Not used.	N/A	N/A	N/A	N/A
08-08 LG	The specific pump discharge pressure value would be moved to the Inservice Testing Program.	Yes	No, containment spray pump parameters were moved to the TRM in LA 37/23.	Yes	Yes
08-09 LG	Moves the requirement that the 18 month verification of automatic containment spray actuation and containment spray pump actuation be performed during shutdown (plant outage) to the Bases.	No, CTS does not require during shutdown.	No, CTS does not require during shutdown.	Yes	Yes

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
08-10 A	ITS Condition F. specifies two containment spray trains or any combination of three or more trains inoperable to enter 3.0.3. Even though this condition is not specified in the CTS, 3.0.3 would be entered.	Yes	No, CPSES has only two containment spray trains covered by this specification. Loss of both of these trains is outside the CTS and 3.0.3 is automatically invoked.	Yes	Yes
08-11 LS2	A "from discovery of failure to meet the LCO provision" has been added to the Completion Time for one train of containment spray/cooling systems inoperable. This change is considered less restrictive in that the 10 days allowed in the ITS not to meet the LCO is greater than the CTS would allow.	Yes	No, CPSES CTS does not have a curtailment cooler specification.	Yes	Yes
09-01 A	The units for the spray additive tank volume limits are changed from gallons to percent.	Yes	No	No	No
09-02 LG	The OPERABILITY of the spray additives educators is contained within the definition of OPERABILITY for the spray additive system as described in the Bases.	Yes	Yes	Yes	Yes
09-03 A	This change revises the ACTION statement by replacing the reference to restoring the spray additive system to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the following 30 hours, with the requirement to be in COLD SHUTDOWN within 78 hours.	Yes	Yes	Yes	Yes
09-04 A	This change adds the phrase "that is not locked, sealed, or otherwise secured in position" with regard to which valves require actuation testing.	Yes	Yes	No, current practice per CTS SR 4.6.2.2.	No, current practice per CTS SR 4.6.2.2.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-05 TR1	The specific actuation signal (a [SI] actuation test signal) for the surveillance was replaced with a generic words that allow credit for an actual or simulated actuation.	Yes	Yes	Yes	No
09-06 LG	This change removes specific details in the SR with regard to verifying flow path and the RWST water flow rates of between 50 and 100 gpm through the eductor test loops, and adds a general requirement to verify flow capability through each eductor.	No, CTS does not contain this detail.	Yes, surveillance details are moved to the Bases.	Yes, surveillance details are moved to the Bases.	No, CTS does not have this system.
09-07 M	The surveillance for DCPD is modified to require demonstration of flow through each solution flow path.	Yes	No	No	No
10-01 LG	Moves details regarding the number of fans in each cooling system train to the Bases.	No, CTS based on different design.	No, CPSES does not have this specification.	Yes	Yes
10-02 LG	Details regarding the automatic functions to be tested and the cooling water flow rate would be moved to the Bases.	No, CTS does not have this detail.	No, CPSES does not have this specification.	Yes	Yes
10-03 LG	The DCPD specific Note cautioning that CFCU flow rate may not be achieved during Section XI testing and RHR operation is relocated to the Bases.	Yes	No	No	No
11-01 LS13	This Note is added to the ACTIONS to allow containment isolation valves that are required to be closed, [] to be opened intermittently under administrative controls.	Yes	Yes	Yes	Yes
11-02 A	This Note is added to the ACTIONS to allow separate Condition entry for each penetration flow path.	Yes	Yes	Yes	Yes
11-03 A	This Note is added to the ACTIONS to enter applicable Conditions and Required ACTIONS for systems made inoperable by containment isolation valves.	Yes	No, a cautionary note is already part of CTS.	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.6

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
11-04 A	NUREG-1431 adds a new Condition to the current Containment Isolation Valve TS to address the case where two containment isolation valves in a penetration flow path are inoperable.	Yes	Yes	Yes	Yes
11-05 LS14	A new Condition is added to the current Containment Isolation Valve TS to cover the case where one containment isolation valve is inoperable in a penetration flow path of the type configured with only one containment isolation valve and a closed system. Also, the Completion Time for this Condition allows 72 hours to isolate an inoperable isolation valve associated with a closed system.	Yes	Yes	No, Wolf Creek does not use GDC 57 valves.	No, Callaway does not use GDC 57 valves.
11-06 TR3	The CTS SR to demonstrate the OPERABILITY of each containment isolation valve by performance of a cycling and isolation time test prior to returning the valve to service after maintenance, repair, or replacement work on the valve or its associated actuator, control, or power circuit has been deleted.	Yes	Yes	Yes	Yes
11-07 LG	The descriptive material regarding the required containment isolation valve actuation signals in the CTS SR is moved to the Bases.	Yes	Yes	Yes	Yes
11-08 TR1	The actuation surveillance is revised consistent with NUREG-1431 to clarify that an actual signal as well as a test signal may be used to verify actuation.	Yes	Yes	Yes	Yes
11-09 A	The isolation time surveillance is revised to delete the reference to verifying "each power operated" containment isolation valve and only require verification of each "automatic isolation valve."	Yes	Yes	Yes	Yes
11-10 A	The note providing a one time test interval extension that is no longer applicable is deleted.	No, CTS does not contain this Note.	Yes	No, CTS does not contain this Note.	No, CTS does not contain this Note.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
11-11 A	A note is added to the containment isolation specification that the LCO is not applicable to MSSVs, MSIVs, FIVs, [] and ADVs.	Yes	No, already part of CTS.	Yes	Yes
11-12 A	The phrase "flow path" is added for clarification and consistency with NUREG-1431.	Yes	Yes	Yes	Yes
11-13 LS22	This change revises the DCPD containment ventilation isolation valve surveillance frequency from 30 months to every 184 days and from 24 hours to 92 days after opening a valve.	Yes	No	No	No
11-14 A	The phrase "that is not locked, sealed, or otherwise secured in position" is added for clarification in regard to which valves require isolation time testing.	Yes	Yes	Yes	Yes
11-15 A	A Callaway specific note to 3.6.3 regarding testing is deleted based on ITS LCO 3.0.5.	No	No	No	Yes
11-16 A	Even though it is not specified in ITS 3.6.3 Required Actions, the ACTION to restore the inoperable valve stated in CTS 3.6.3.a is understood as always the primary objective and a continuous option to be performed during any Completion Time.	Yes	Yes	Yes	Yes
11-17	Not Used.	N/A	N/A	N/A	N/A
11-18 LG	The Callaway specific words "during the COLD SHUTDOWN or REFUELING MODE" are moved to the Bases.	No	No	No	Yes
12-01 A	Consistent with NUREG-1431, the hydrogen monitoring specification is moved to ITS Section 3.3.3, concerning PAM instrumentation.	Yes	Yes	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-02 M	The MODE of Applicability for the hydrogen monitors is extended to MODE 3.	Yes	Yes	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
12-03 LS15	The ACTION is revised to require a special report to be submitted within 14 days in lieu of being in HOT STANDBY within 6 hours, if one train of hydrogen monitoring cannot be restored to OPERABLE within 30 days.	Yes	Yes	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
12-04 M	Adds the requirement to be in HOT SHUTDOWN within 12 hours if both trains of hydrogen monitoring are inoperable and one train was not restored within 72 hours.	Yes	Yes	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
12-05 LS16	Revises the frequency of the surveillance to perform CHANNEL CALIBRATION from 92 days on a staggered basis to once per 18 months.	Yes	No, CTS requirement redefined (see 12-06-LG).	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
12-06 LG	The details provided for performing the CHANNEL CALIBRATION are moved out of the SR. The information is moved to the Bases.	Yes	Yes	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
12-07 M	A new SR is added for DCPD requiring a CHANNEL CHECK every 31 days for the hydrogen analyzer/monitors.	Yes	No, SR already in CTS.	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
13-01 LS17	A new Condition has been added to this specification. This condition describes the Required Action for two hydrogen recombiners inoperable.	Yes	Yes	Yes	Yes
13-02 LS18	The current SR to perform a hydrogen recombiner functional test every 6 months is revised to every 18 months.	Yes, CTS required Refueling Interval changed to 18 months.	Yes	Yes	No, CTS already has 18 month.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
13-03 LG	Descriptive information regarding the current hydrogen recombiner surveillance is moved into the Bases.	Yes	Yes	Yes	Yes
13-04 LG	The SR to perform a Channel CALIBRATION on all the hydrogen recombiner instrumentation is moved to Licensee controlled document.	Yes, moved to an ECG.	Yes, moved to the TRM.	Yes, moved to USAR Chapter 16.	Yes, moved to FSAR Chapter 16.
13-05 LS23	Added statement that LCO 3.0.4 is not applicable to ACTION.	Yes	No, CTS already has the LCO 3.0.4 not applicable statement.	Yes	Yes

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR1, 2, 3...." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1

10CFR50.92 EVALUATION
FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A Note is added to valve and blind flange SRs consistent with NUREG-1431. The note allows verification of valves, flanges, and isolation devices located in high radiation areas to be verified by use of administrative means. This change is less restrictive in that the CTS SR 4.6.1.1 has an exception to valves, blind flanges, and deactivated automatic valves which are located inside containment and are locked, sealed, or otherwise secured in the closed position. These valves shall be verified closed during each COLD SHUTDOWN. However, under the CTS, if an area outside of containment became a high radiation area, entry into the area would still be required to verify the closed positions. The ITS would allow verification of all areas that are high radiation areas or become high radiation areas by administrative means once they had been verified to be in the proper position. This is considered acceptable, since access to these areas is restricted for ALARA reasons. Therefore, the probability of misalignment of these devices, once they have been initially verified in the proper position, is small.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves upgrading the Containment Isolation Valve TS to more closely agree with the Westinghouse Standard ITS (NUREG-1431) and does not result in any hardware changes. The isolation devices affected are not assumed to be an initiator of any analyzed event. The isolation devices are passive and serve to limit the consequences of accidents. The proposed change ensures the isolation devices remain positioned to limit the consequences of design basis events as described in the Final Safety Analysis Report (FSAR) and that the results of the analyses in the FSAR remain bounding. Additionally, the proposed change does not impose any new safety analyses limits or alter the plant's ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves upgrading the Containment Isolation Valve TS to more closely agree with the Westinghouse Standard TS (NUREG-1431) and does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change, which upgrades the Containment Isolation Valve TS to be consistent with the Westinghouse Standard TS (NUREG-1431), does not involve a significant reduction in a margin of safety. The proposed change has been developed considering the importance of the isolation devices in limiting the consequences of a design basis event and ALARA concerns for the plant personnel. The proposed change and ALARA access restrictions still ensure the isolation devices are properly positioned to limit the consequences of a design basis event. In addition, the proposed change eliminates unnecessary exposure of plant personnel to high radiation areas.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS1" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A provision, "within 10 days from discovery of failure to meet the LCO," has been added to the Completion Time for the ACTIONS with the "Containment Cooling System" inoperable. The CTS, "Containment Spray," and "Containment Cooling," systems require restoring the inoperable system to OPERABLE status within 72 hours. The CTS limits the inoperability of any combination of these two systems to 72 hours or it provides a maximum of 7 days for restoring one group of cooling fans to OPERABLE status when everything else is OPERABLE. NUREG-1431 provides a maximum of 10 days for not meeting the LCO. The 10 day provision in the Completion Time is considered appropriate based upon engineering judgment considering the low probability of coincident entry into two Conditions in this specification coupled with the low probability of an accident occurring during this time.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves a provision, "within 10 days from discovery of failure to meet the LCO," being added to the Completion Time for the ACTIONS with the "Containment Cooling System" inoperable. The CTS, "Containment Spray," and "Containment Cooling," systems require restoring the inoperable system to OPERABLE status within 72 hours. The CTS limits the inoperability of any combination of these two systems to 72 hours or it provides a maximum of 7 days for restoring one group of cooling fans to OPERABLE status when everything else is OPERABLE. NUREG-1431 provides a maximum of 10 days for not meeting the LCO. The 10 day provision in the Completion Time is considered appropriate based upon engineering judgment considering the low probability of coincident entry into two Conditions in this specification coupled with the low probability of an accident occurring during this time. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation.

The proposed change will make a provision for a maximum 10 day allowance for not meeting the LCO. The provision takes into account the capability of the remaining systems based upon the applicable Conditions entered. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2 (continued)

3. Does this change involve a significant reduction in a margin of safety?

Analysis shows that [one train of the containment cooling together with one train of containment spray] can provide 100 percent of the required peak cooling capacity during the post accident Conditions. The provision to allow 10 days not meeting the LCO for one containment spray train inoperable, or one train of the required containment cooling trains inoperable, is acceptable taking into account the low probability of coincident entry into two Conditions coupled with the low probability of an accident occurring during this time. The provision also takes into account the capability of the remaining systems based upon the applicable Conditions entered. Thus, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS2" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, a Note has been added to the Containment Airlock TS to allow entry and exit into containment via the air locks if both air locks are inoperable and administrative controls are provided. This is permitted for up to 7 days. With both air locks inoperable, containment entry may be required on a periodic basis to perform TS surveillances and Required ACTIONS, as well as other activities on equipment inside containment. This Note provides allowance for these activities to be performed.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves changing the Containment Air Lock TS to more closely agree with the Westinghouse Standard TS (NUREG-1431) and does not result in any hardware changes. The containment air lock is not assumed to be the initiator of any analyzed event. The role of the containment air lock is in containing releases to the containment during a design basis accident (DBA), and thereby limiting consequences. The requested change does not allow continuous operation such that a single failure could allow a release from containment during a DBA. The proposed allowance would only apply for 7 days. Therefore, the likelihood of the door being open when an accident occurs is very small. Additionally, during the period when this allowance applies, entry and exit is supervised by personnel such that in the event of an incident or unusual condition the operable air lock door could be quickly closed, thereby reestablishing the containment boundary. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves changing the Containment Air Lock TS to more closely agree with the Westinghouse Standard TS (NUREG-1431) and does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure the containment boundary is capable of being maintained. Since air lock malfunction is not the initiator of any event, having the air lock open for a brief period during plant operation could not result in the occurrence of a new or different type of accident. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change, which revises the Containment Air Lock TS to be consistent with the Westinghouse Standard TS (NUREG-1431), does not involve a significant reduction in a margin of safety. The allowance to permit entry and exit under administrative controls is acceptable based on the small probability of an event requiring the containment air lock to mitigate consequences of DBAs occurring while the air lock is open. The requested change provides the ability to repair an inoperable air lock door or enter containment to perform surveillances with both air locks inoperable. The exposure of the plant to the small probability of an event requiring the operable containment air lock to be closed during the short time period it is opened to permit entry and exit under administrative controls is insignificant. Additionally, this change provides the benefit of avoiding an unnecessary plant shutdown transient by allowing entry and exit to perform surveillances or Required Actions.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS3" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The allowance to continue operation with one air lock door inoperable has been modified to remove the restriction which limits this condition until the next overall air lock leakage test. This restriction was removed because the air lock remains capable of performing its safety function with the remaining operable door. Therefore, continued operation may proceed indefinitely subject to the other restrictions of the TS (continuing to meet the ACTIONS and applicable surveillances). This change is consistent with NUREG-1431.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes. An inoperable air lock door is not assumed to be an initiator of any accident previously evaluated. The change allows operation to continue if the overall air lock leakage test passes since the containment boundary is maintained intact in this Condition by the remaining operable door. Administrative controls are implemented to provide an interlock function such that containment integrity is not jeopardized. As such, no increase in the probability or consequences of an accident is involved with this change.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will provide an allowance to continue operations indefinitely with one air lock door inoperable subject to the other restrictions of the specification. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

No reduction in a margin of safety is involved since the containment boundary remains intact and compensatory administrative controls are invoked such that there is no adverse impact created by this change. In addition, this change provides the benefit of avoiding an unnecessary plant transient required by TS when overall leakage is still within limits.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4
(continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS4" resulting from the conversion to the ITS format satisfy the NSCH standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, this note modifies the requirement to verify an air lock door locked closed every 31 days. The note allows the verification of locked closed air lock doors located in high radiation areas to be performed by use of administrative means. This change is less restrictive in that the CTS does not allow this exception due to current design and capability to verify the inner door locked from outside the containment airlock. Under the CTS, if an area outside of the airlock became a high radiation area, entry into the area would still be required to verify the closed positions. The ITS would allow verification of all areas that are high radiation areas or become high radiation areas by administrative means once they had been verified to be in the proper position. This is considered acceptable, since access to these areas is restricted for ALARA reasons. Therefore, the probability of misalignment of the air lock doors once they have been initially verified in the proper position, is small.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves changing the Containment Air Lock TS to more closely agree with the Westinghouse Standard TS (NUREG-1431) and does not result in any hardware changes. The air locks are not assumed to be an initiator of any analyzed event. The air locks are passive and serve to limit the consequences of accidents. The proposed change still ensures the air locks remain closed to limit the consequences of design basis events as described in the FSAR and that the results of the analyses in the FSAR remain bounding. Additionally, the proposed change does not impose any new safety analyses limits or alter the plant's ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves changing the Containment Air Lock TS to more closely agree with the Westinghouse Standard TS (NUREG-1431) and does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The air locks are not the initiators of any event. Allowing verification of the position of the operable air lock door by administrative means if the door is located in a high radiation area could not result in the occurrence of a new or different kind of accident. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change, which revises the Containment Air Lock TS to be consistent with the Westinghouse Standard TS (NUREG-1431), does not involve a significant reduction in a margin of safety. The proposed change has been developed considering the importance of the air locks in limiting the consequences of a design basis event and ALARA concerns for the plant personnel. The proposed change and ALARA access restrictions still ensure the isolation devices are properly positioned to limit the consequences of a design basis event. In addition, the proposed change eliminates unnecessary exposure of plant personnel to high radiation areas.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS5" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, a Condition and associated Required Actions have been added which allows continued operation when the personnel air lock is inoperable due to an inoperable interlock mechanism. Use of the air lock is permissible under the control of a dedicated individual since an equivalent level of assurance that only one door will be open at a time is provided. The ACTIONS and modifying notes provided for the interlock mechanism are consistent with the ACTIONS and notes for an inoperable air lock door. These ACTIONS ensure that an operable door is closed and locked and entry or exit is controlled by a dedicated individual. While this change provides a slightly less restrictive TS requirement, the overall impact on plant safety is negligible due to the Required Actions implemented.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes. The air lock door interlock mechanism is not assumed to be an initiator of any accident previously evaluated. The function of the air lock door mechanism is to prevent simultaneous opening of both doors in a single air lock, thus providing a release path from containment to the environment. Both the inner and outer doors of an air lock are designed to withstand the maximum expected post-accident containment pressure. During the time when the air lock door interlock mechanism is inoperable, and ingress and egress of containment is necessary, the use of a dedicated individual is provided to ensure only one air lock door is opened at a time. This method of control is reasonable since an equivalent level of assurance is provided that one door in a single air lock will remain closed, thereby ensuring that containment remains operable. The consequences of an event occurring during the indefinite operation with the air lock door interlock mechanisms inoperable are the same as the consequences of an event occurring during the current allowance of 24 hours. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Since the air locks are not the initiators of any event and serve only to mitigate the consequences of events, this change could not result in the occurrence of a new or different kind of accident. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed allowance to use a dedicated individual to ensure one air lock door in a single air lock is maintained closed when the air lock door interlock mechanism is inoperable does not have any effect on accident or transient analyses. As a result, the proposed change deletes the requirement to place the plant in MODE 3 when the air lock door interlock mechanism is inoperable. This is considered acceptable since the dedicated individual serves the function of the interlock mechanism to ensure one air lock door is always maintained closed. The closed single air lock door is designed to withstand the maximum expected post-accident containment pressure, and thereby ensures the assumptions of the applicable safety analysis are maintained. As such, any reduction in a margin of safety created by allowing the air lock door interlock mechanism to be inoperable for an indefinite time will be offset by the benefits gained by avoiding an unnecessary plant transient when an adequate compensatory measure exists.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS6" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, a Note is added to allow unrestricted access (entry and exit) to repair air lock components in an inoperable air lock. The ACTION requirement to close and lock an air lock door remains applicable with the exception of the brief entry and exit of the air lock to perform necessary repairs. The allowance to enter and exit through the air lock door required closed by the ACTIONS is acceptable based on the low probability of an event occurring that could challenge the containment boundary during the short time the door is open for entry and exit. In addition, this allowance reduces the potential risk incurred during a plant transient that may result from a shutdown required by TS due to the inoperable air lock.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves changing the Containment Air Lock TS to more closely agree with the Westinghouse Standard TS (NUREG-1431) and does not result in any hardware changes. The containment air locks are not assumed to be the initiator of any analyzed event. The containment air locks are assumed closed in the safety analyses to limit the consequences of accidents. The proposed change allows the locked closed door to be opened briefly to facilitate entry and exit for repairs. The brief time the door is open for entry or exit is acceptable based on the extremely low probability of an event occurring that would challenge the containment boundary in the brief time it takes to enter or exit through the air lock door. The proposed change does not alter the intent of the TS requirement which, in effect, still ensures that the containment air locks will remain closed to limit the consequences of design basis events as described in the FSAR and that the results of the analyses in the FSAR remain bounding. Additionally, the proposed change does not impose any new safety analyses limits or alter the plant's ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves changing the Containment Air Lock TS to more closely agree with the Westinghouse Standard TS (NUREG-1431) and does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change, which revises the Containment Air Lock TS to be consistent with the Westinghouse Standard TS (NUREG-1431), does not involve a significant reduction in a margin of safety. The proposed change allows the locked closed air lock door to be opened briefly to facilitate entry and exit for repairs. The brief time the door is open for entry or exit is acceptable based on the extremely low probability of an event occurring that would challenge the containment boundary in the brief time it takes to enter or exit through the air lock door. Therefore, the proposed change does not alter the intent of the TS requirement, which in effect, still ensures that the containment air locks will remain closed to limit the consequences of design basis events as described in the FSAR and that the results of the analyses in the FSAR remain bounding. In addition, the proposed change reduces the potential risk incurred during a plant transient that may result from a shutdown required by TS due to the inoperable air lock. Therefore, any reduction in the margin of safety is insignificant and offset by the reduction in potential plant transients.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS7" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with industry Traveler TSTF-17, Rev. 1, this change would extend the testing frequency for the containment airlock interlock mechanisms from 6 months to 24 months.

Air lock interlock mechanisms are typically enabled after each refueling outage, then verified OPERABLE with this surveillance, and not disturbed until the next refueling outage. Should the need for maintenance on an interlock mechanism arise while the interlock is required, performance of this surveillance would be required following the interlock maintenance. In addition, when an air lock is opened during the time the interlock is required, the operator first verifies that one door is completely shut before attempting to open the other door. Therefore, the interlock is not normally challenged except during actual testing of the interlock. Consequently, testing of the interlock on a 24 month interval is sufficient to ensure its proper operation.

Testing of the air lock mechanism is accomplished through having one door open, then attempting to open the second door. Failure of the interlock as a result of this surveillance would effectively result in a momentary loss of containment integrity until one door could be reclosed. The air lock interlock mechanism is not challenged during ingress and egress based on training, procedures, and conservative operating practices. One door is opened, all personnel and equipment as necessary are placed into the air lock, and then the door is completely closed prior to attempting to open the second door. Performance of this surveillance when the interlock function is required is contrary to conservative operating practices and training. In addition, failure rates of the interlock mechanisms are very low due to their design. Also, the door interlock mechanisms cannot be readily bypassed and are under the control of station administrative processes. By adopting a 24 month interval for the required testing frequency, the number of challenges to containment integrity due to interlock testing would be reduced.

Historically, the interlock verification frequency has coincided with the frequency of the overall air lock leakage test, which in accordance with 10 CFR 50, Appendix J, Option A, is once per 6 months. With the adoption of Appendix J, Option B, the overall air lock leakage test frequency requirement is 30 months. By changing the required interlock test surveillance frequency to 24 months the interlock testing frequency would correspond to the overall air lock leakage test frequency allowable under Option B.

The 24 month interlock frequency is based on engineering judgement and is considered adequate given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8 (Continued)

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change extends the testing frequency of containment interlock mechanisms from 6 months to 24 months. This change is consistent with the change proposed to the Westinghouse TS (NUREG-1431) by industry traveler TSTF-17, Rev. 1. The air lock does not serve as an initiator of an accident nor does this change result in any hardware changes. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change extends the testing frequency of containment air lock interlock mechanisms from 6 months to 24 months. The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Since the air locks are not the initiators of any event and serve only to mitigate the consequences of events, this change could not result in the occurrence of a new or different kind of accident. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. The interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit, nor is it challenged during an accident. By adopting a 24 month interval for the required frequency the number of challenges to containment integrity due to interlock testing would be reduced. None of these changes impact the margin of safety in any way. Thus there is no reduction in the margin of safety from that previously established.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS8" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The required ACTION for a containment ventilation isolation valve not within its leakage limit is revised to allow the penetration to be isolated using a closed and deactivated automatic valve, a closed manual valve or a blind flange and does not require the isolation valve to be restored to OPERABLE status. This is an option not explicitly available in the CTS. The Completion Time of 24 hours remains the same as in the CTS. If valves with resilient seals are used to isolate the flow path, the leakrate of these valves must be verified at least every 92 days. If a blind flange is used to isolate the penetration flow path, the valves with resilient seals whose flow is isolated by the blind flange are not required to be leakrate tested. Isolation of the flow path with a leak tested blind flange provides the required leak barrier and additional leak testing of the valves in the flow path is redundant and unnecessary.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes. The role of these valves during an accident is to isolate containment, and thereby limit accident consequences. The proposed ACTIONS will not allow continuous operation such that containment leakage after an accident will exceed assumed values. With the associated penetrations isolated per the proposed ACTION requirements, no accident as a result of the leaking valve is credible. Further, with the line isolated it cannot contribute to the consequences of a previously evaluated accident. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change is acceptable since the isolated penetration provides protection equivalent to restoring the valve to OPERABLE status. Providing the option to isolate the penetration will minimize the potential for plant transients that could occur during a shutdown required by TS if the isolation valve could not be restored to OPERABLE status. In addition, the isolation of the line (in accordance with the proposed Required Action) ensures that leakage through the associated penetration is within limits. As such, any reduction in a margin of safety will be insignificant and offset by the benefit gained through avoiding an unnecessary plant transient.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS9" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS13
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A Note is added to the ACTIONS in accordance with the NUREG-1431, LCO 3.6.3, Containment Isolation Valves which provides an allowance to open, under administrative controls, containment isolation valves required to be closed []. This is acceptable based on administrative controls consisting of a dedicated operator at the valve in continuous communication with the control room. These controls provide for the capability to manually close the isolation valve should an automatic isolation be required. []

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves changing the Containment Isolation Valve TS to more closely agree with the Westinghouse Standard TS (NUREG-1431). The change does not result in any hardware changes. The isolation valves act to isolate the containment penetrations in the event of a DBA and serve to limit the consequences of accidents. The proposed change continues to ensure that the isolation valves will perform their required function and limit the consequences of design basis events as described in the FSAR and that the results of the analyses in the FSAR remain bounding. The proposed change does not impose any new safety analyses limits or alter the plant's ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves changing the Containment Isolation Valve TS to more closely agree with the Westinghouse Standard TS (NUREG-1431) and does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS13 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change revises the Containment Isolation Valve TS to be consistent with the Westinghouse Standard TS (NUREG-1431) and does not involve a significant reduction in a margin of safety. The proposed change has been developed considering the importance of the containment isolation valves in limiting the consequences of a design basis event and the concerns for the plant's ability to perform required operational support functions with necessary systems isolated. The proposed change allows for protection commensurate with that provided by an automatic isolation system. Considering the probability of an event that would challenge the containment boundary, the alternative protection provided by this change and the operational requirements to occasionally open these valves, the proposed change is acceptable and any reduction in the margin of safety insignificant.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS13" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14
10 CFR 50.92 EVALUATION
FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A new Condition and Required Actions consistent with NUREG-1431 and modified in accordance with TSTF-30, Rev. 1, is added to the current Containment Isolation Valve TS to cover the case where one containment isolation valve is inoperable in a penetration flow path of the type configured with only one containment isolation valve and a closed system. GDC 57 allows the use of a closed system in combination with a containment isolation valve to provide the two containment barriers against the release of radioactive material following an accident. The CTS LCO 3.6.3 does not allow the use of a closed system to isolate a failed containment isolation valve even though the closed system is subjected to a Type A containment leakage test, is missile protected, and is seismic Category I piping. A closed system also typically has flow through it during normal operation such that any loss of integrity could be continually observed through leakage detection systems within containment and system walkdowns for closed systems outside containment. As such, the use of a closed system is no different from isolating a failed containment isolation valve by use of a single valve as specified in NUREG-1431 Required Action A.1. Therefore, in accordance with NUREG-1431 and Traveler TSTF-30, Rev. 1, the Required Action for this condition allows 72 hours to isolate a failed valve associated with a closed system. 72 hours provides the necessary time to perform repairs on a failed containment isolation valve when relying on an intact closed system. A Completion Time of 72 hours is considered appropriate given that certain valves may be located inside containment, the reliability of the closed system, and that 72 hours is typically provided for losing one train of redundancy throughout NUREG-1431. If the closed system and associated containment isolation valve were both inoperable, the plant would be in LCO 3.0.3 since there is no specific Condition specified.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Consistent with NUREG-1431 as modified in accordance with industry Traveler TSTF-30, Rev. 1, the proposed change adds a new Condition and Required Actions to the Containment Isolation Valves specification for the case where one containment isolation valves is inoperable in a penetration flow path of the type configured with only one containment isolation valve and a closed system. GDC 57 allows the use of a closed system in combination with a containment isolation valve to provide two containment barriers. This change does not result in any hardware changes or alter the plant's ability to detect and mitigate events. Since containment isolation is an accident mitigating function and not an initiator of an accident, it can not increase the probability of an accident. The use of a closed system is no different from isolating a failed containment isolation valve by use of a single valve. A completion time of 72 hours is appropriate given that certain valves may be located inside containment, the reliability of the closed system, and that 72 hours is typically provided for losing one train of redundancy throughout the NUREGs. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation, and does not impose any new safety analyses limits. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change adds a new Condition and Required Actions to the Containment Isolation Valves specification to be consistent with the Westinghouse Standard TS (NUREG-1431) and does not involve a significant reduction in a margin of safety. The proposed change has been developed considering the importance of the containment isolation valves in limiting the consequence of a design basis event and the reasonable time to perform repairs on a failed containment isolation valve when relying on an intact closed system. Considering the probability of an event that would challenge the containment boundary and the reliability of the closed system, the proposed change is acceptable and any reduction in the margin of safety would be insignificant. As such any reduction in margin of safety will be insignificant and offset by the benefit gained through avoiding an unnecessary plant transient.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS14" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly, a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS15
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The requirements for the hydrogen monitors are combined into the new PAM TS and revised to be consistent with other PAM instrumentation. The current containment hydrogen monitor ACTION statements are replaced by the specified NUREG-1431 Conditions. These Conditions are based on the required channels operable column. The 30 day Completion Time allowed by the NUREG-1431 is based on operating experience and takes into account the remaining OPERABLE channel and the function of the instrument (no critical automatic action is initiated from this instrumentation). The requirement to submit a special report in lieu of a plant shutdown is appropriate since alternative ACTIONS are identified before a loss of functional capability occurs and given the likelihood of unit conditions that would require information provided by this instrumentation. These changes are consistent with NUREG-1431.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes affect instrumentation that would be used to assess the condition of the plant during and following an accident. As such, the changes can have no effect on the probability of any accident previously evaluated since the hydrogen monitors have no bearing on initiating events. The proposed changes will continue to ensure the capability to monitor plant conditions during and following an accident by requiring redundancy or diversity and timely corrective action in the event of inoperable instrumentation. Therefore, the proposed changes will not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes affect the operability and action requirements for the PAM instrumentation system. Accordingly, the proposed changes do not involve any change to the configuration or method of operation of any plant equipment, and no new failure modes have been defined for any plant system or component nor has any new limiting failure been identified as a result of the proposed changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS15 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed changes do not involve a significant reduction in a margin of safety. The intent of the existing TS requirements is to ensure the capability to monitor the plant condition during and following an accident so that the operators will have the information necessary to monitor and evaluate the course of the event and take any necessary action. Under the proposed changes this capability will be maintained by ensuring redundancy or diversity and by requiring timely corrective action in the event of inoperable instrumentation. In addition, the proposed changes would avoid unnecessary plant shutdowns by specifying an appropriate level of action in response to inoperable instrumentation. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS15" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Revises the frequency of the hydrogen monitor surveillance to perform CHANNEL CALIBRATION from 92 days on a staggered basis to once per 18 months consistent with NUREG-1431. The hydrogen monitors are part of the PAM instrumentation and their primary function is to provide information required by the control room operator during accident situations. The hydrogen monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The hydrogen monitors provide information required by the control room operator during accident situations and their failure would not increase the probability of an accident previously evaluated.

The failure of the hydrogen monitors could potentially increase the consequences of an accident previously evaluated if hydrogen were allowed to accumulate during an accident situation. However, the primary means of reducing hydrogen concentration during accidents is via the independent hydrogen recombiners []. Failure of the monitors would not affect the capabilities of that system. Further, the extension of the CHANNEL CALIBRATION surveillance interval from 92 days (on a staggered test basis) to every 18 months is not expected to effect the reliability or performance of the hydrogen monitors based on industry operating experience. Also, this surveillance interval is consistent with frequency of other PAM instrumentation. Therefore the extension of the surveillance interval would not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change affects the frequency of a SR for the PAM instrumentation system. It does not involve any change to the configuration or method of operation of any plant equipment, and no new failure MODES have been defined for any plant system or component, nor has any new limiting failure been identified as a result of the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16 (continued)

3. Does this change involve a significant reduction in a margin of safety?

The accident analyses are assumed to be initiated from Conditions which are consistent with the TS LCO. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no effect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS16" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS17
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A new Condition and Required Action have been added to this specification. This Condition describes the Required Action for two hydrogen recombiners inoperable. This change allows up to 7 days to restore one hydrogen recombiner to OPERABLE status, based on the availability of the containment hydrogen purge system to provide the required safety function. In order to utilize this action time, the Required Actions require that the hydrogen control function be verified available within 1 hour and once every 12 hours thereafter. This administrative verification will assure that other means of controlling hydrogen, e.g., hydrogen purge, are capable of performing the safety function if an event occurs. Also, the Bases for operation of the recombiners indicates that if a design basis event occurs, 8 or more days would elapse before the containment atmosphere approached the lower flammability limit for hydrogen. It is, therefore reasonable to assume that the inoperability of two hydrogen recombiners will not significantly jeopardize the capability of the facility to respond to a design basis event. This change is consistent with NUREG-1431.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change does not result in any hardware changes. The hydrogen recombiners are not assumed to be initiators of any analyzed event. The proposed change allows two hydrogen recombiners to be inoperable for up to 7 days provided an alternate hydrogen control function is available. The alternate hydrogen control capability is provided by the containment hydrogen purge system. This method of hydrogen control is acceptable since the hydrogen purge system is capable of maintaining the hydrogen concentration in containment below flammability limit, thus ensuring the pressure and temperature assumed in the applicable safety analysis are not exceeded. A 1 hour Completion Time to initially verify the availability of the alternate hydrogen control function and additional verification once per 12 hours thereafter, is an acceptable frequency to ensure the alternate hydrogen control function is maintained. The proposed Completion Time of 7 days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of a DBA, which would generate hydrogen in the amount capable of exceeding the flammability limit. While the hydrogen purge system maintains the hydrogen concentration below the lower flammability limit during a loss-of-coolant accident (LOCA), its operation is fundamentally different than the operation of the hydrogen recombiners. The hydrogen recombiners process containment atmosphere entirely within the containment (i.e., suction and discharge remain in the containment), whereas the hydrogen purge system takes suction on the containment atmosphere, processes it outside containment via atmospheric cleanup units (particulates, iodine absorbers, and HEPA filters), and exhausts the cleansed air through the vent stacks.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS17 (continued)

Vent stack radiation monitors help to ensure the exhaust remains within discharge limits. Outside air replaces the exhaust air to maintain containment pressure. The operation of the hydrogen purge system allows some gaseous radioactive products to escape the containment (e.g., noble gases), and may increase the potential of an inadvertent radioactive release. However, the design of the hydrogen purge system and testing requirements, as well as the vent stack monitoring ensure that operation of the system does not significantly increase the potential for releases in excess of accidents previously evaluated. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change will allow two hydrogen recombiners to be inoperable for up to 7 days provided an alternate hydrogen control function is maintained. The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The allowance to have two hydrogen recombiners inoperable for up to 7 days, provided an alternate hydrogen control function is available, does not have any effect on accident or transient analysis. Use of the hydrogen purge system as the alternate means of hydrogen control ensures the temperature and pressure assumptions in the applicable safety analysis are maintained. The 1 hour Completion Time to initially verify the availability of the alternate hydrogen control function, and the additional verification once per 12 hours thereafter, is an acceptable frequency to ensure availability of the hydrogen purge system is maintained. The 7 day Completion Time to restore the inoperable hydrogen recombiners prior to requiring a plant shutdown is acceptable based on the small probability of an event requiring the hydrogen recombiners to function during this time period and the availability of the hydrogen purge system to control hydrogen concentration in containment. Providing the 7 day Completion Time when two hydrogen recombiners are inoperable will minimize the potential for plant transients that can occur while shutting down the plant by providing an adequate time to restore an inoperable hydrogen recombiner to operable status. As such, any reduction in a margin of safety by allowing two hydrogen recombiners to be inoperable for 7 days will be offset by the benefits gained by avoiding an unnecessary plant transient by providing adequate time to restore a hydrogen recombiner to OPERABLE status when the hydrogen control function can still be satisfied.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS17" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS18
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The current surveillance requirement to perform a hydrogen recombiner functional test every 6 months is revised to every 18 months consistent with NUREG-1431. This change is considered acceptable due to the redundancy and proven high reliability of the system. Hydrogen recombiner operating experience has shown that functional test failures are rare. In addition, the fully redundant and independent hydrogen purge system provides an alternate, and equally effective, method of controlling hydrogen. The proposed change is in accordance with NUREG-1366, "Improvement to Technical Specification Requirements," and NUREG-1431.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change, to extend the surveillance interval for the hydrogen recombiner functional tests, does not result in any hardware changes. The hydrogen recombiners are not assumed in the initiation of any analyzed event. Their role is in reducing hydrogen concentration in containment, and thereby limiting potential accident consequences. Hydrogen recombiner operating experience has shown that functional test failures are rare. Thus, the extended surveillance interval will not result a loss in the capability to reduce hydrogen concentration in containment. Additionally, in the unlikely event both recombiners fail, a diverse method of reducing hydrogen concentration is available utilizing the containment purge system.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure the recombiners are maintained operable. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The increased interval between hydrogen recombiner functional tests is acceptable based on the relative simplicity of the recombiner system and industry experience which indicates that the recombiner availability can be assured with reduced testing. As a result, any reduction in a margin of safety will be insignificant.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS18
(continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS18" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS19
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change would specify that containment isolation valves which are locked, sealed, or otherwise secured are not required to be verified closed during periodic surveillances since these were verified to be in the correct position prior to locking, sealing, or securing. If locked, sealed, or otherwise secured manual valves, blind flanges, and deactivated automatic valves are closed to satisfy an ACTION (e.g., 3.6.3) the position must be verified but may be verified by administrative means. This change is consistent with NUREG-1431 requirements for other valves required to be in the correct position prior to an accident, i.e. emergency core cooling system (SR 3.5.2.2), auxiliary feedwater (SR 3.7.5.1), and SW (SR 3.7.8.1) and is in accordance with industry Traveler TSTF-45 and WOG-91. The probability of misalignment of these devices, once they have been initially verified in the proper position, is small.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change, to not require that containment isolation valves which are locked, sealed, or otherwise secured be verified closed during periodic surveillance (or to verify their closure was required by a required action), involves upgrading the Containment Isolation Valve TS to more closely agree with the Westinghouse Standard ITS (NUREG-1431) and does not result in any hardware changes. The isolation devices affected are not assumed to be an initiator of any analyzed event. The isolation devices are passive and serve to limit the consequences of accidents. The proposed change would still ensure that the isolation devices remain positioned to limit the consequences of design basis events as described in the FSAR and that the results of the analyses in the FSAR remain bounding. Additionally, the proposed change does not impose any new safety analyses limits or alter the plant's ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration to the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change would still ensure that the isolation devices remain positioned to limit the consequences of design basis events as described in the FSAR and that the results of the analyses in the FSAR remain bounding. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS19 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

This change would specify that containment isolation valves which are locked, sealed, or otherwise secured are not required to be verified closed during periodic surveillance since these were verified to be in the correct position prior to locking, sealing, or securing. The proposed change still ensures that the isolation devices are properly positioned to limit the consequences of a design basis event. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS19" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS22
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change revises the containment ventilation isolation valve leakage surveillance frequency after opening a valve from 24 hours to 92 days. This change is consistent with NUREG-1431. These valves have a good service record and have consistently met leakage rate requirements. The revised 92 day frequency still reflects conservative margin to compensate for degradation of the resilient seats in these valves. This change also adds the more restrictive change that the valves be leak tested every 184 days (CTS states 18 months) even if they have not been used.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92 (c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

These valves fully meet the requirement of the Branch Technical Position CSB 6-4 (except for their larger size). The ten year service records for these valves show that they consistently meet their leakage rate limits. The revised surveillance frequency still retains the NRC recommended adequate margin to compensate for the fact that these valves are resilient seated valves. Therefore, the proposed change to surveillance frequency does not effect the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change to the surveillance frequency does not involve a physical alteration to any plant equipment, causes no change in the method by which any safety related system performs its function, and does not alter the manner in which any safety system is operated. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

These valves fully meet the requirement of the Branch Technical Position CSB 6-4 (except for their larger size). The [nine] year service records for these valves show that they consistently meet their leakage rate limits. The revised surveillance frequency still retains the NRC recommended adequate margin to compensate for the fact that these valves are resilient seated valves. Therefore, there is no significant reduction in the margin of safety.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS22
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS22" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed change meets the requirements of 10 CFR 50.92 (c), and does not involve a significant hazards consideration.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS23
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The current ACTION statement, "with one of the two Hydrogen Recombiner Systems inoperable, restore the inoperable Hydrogen Recombiner System to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours," is being modified by a note that states the provisions of LCO 3.0.4 are not applicable consistent with NUREG-1431, Rev. 1. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes. The hydrogen recombiners are not assumed in the initiation of any analyzed event. In the condition that is being revised, one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status. However, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. The overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The added note, "the provisions of 3.0.4 not applicable," allows a MODE change when one recombiner is inoperable. Exceptions to the requirements of 3.0.4 allow entry into MODES or other specified Conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified Condition in the Applicability only for a limited period of time. Additionally, in the unlikely event both recombiners fail, a diverse method of reducing hydrogen concentration is available utilizing the containment purge system. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS23 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure the recombiners are maintained operable. This change is being made consistent with NUREG-1431 that allows entry into MODES or other specified Conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified Condition in the Applicability only for a limited period of time. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

In the Condition that is being revised, one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status. However, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. The overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. However, in the unlikely event both recombiners fail, a diverse method of reducing hydrogen concentration is available utilizing the containment purge system. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS23" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c) and accordingly a NSHC finding is justified.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR1 10 CFR 50.92 EVALUATION FOR RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the existing TS to additionally allow the use of actual actuation signals for surveillances that currently call for testing using simulated test signals only. This change achieves consistency with the proposed ISTS (NUREG-1431).

In several specifications throughout the TS, OPERABILITY of certain equipment is demonstrated by ensuring that the equipment performs its safety function upon receipt of a simulated test signal. The intent of a 'simulated' signal was to be able to perform the required testing without the occurrence (or without causing) an actual signal generating event. However, the unintended effect was to require the performance of the surveillance (using a test signal) even if an actual signal had previously verified the operation of the equipment. This change allows credit to be taken for actual events when the required equipment actuates successfully.

While the occurrence of events that cause actuation of accident mitigation equipment is undesirable, the actuation of mitigation equipment on an actual signal is a better demonstration of its OPERABILITY than an actuation using a test signal. Thus the change does not reduce the reliability of the equipment tested. The change also improves plant safety by reducing the amount of time the equipment is taken out of service for testing and thereby increasing its availability during an actual event and by reducing the wear of the equipment caused by unnecessary testing.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or
2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or
3. Involve a significant reduction in a margin of safety."

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change allows the use of an actual actuation signals (when/if it occurs) to satisfy SRs currently requiring simulated test signals to demonstrate equipment OPERABILITY. While the change takes advantage of events that may have occurred, it has no adverse effect on any accident initiators or accident consequences. In fact, by potentially reducing unnecessary testing, it may reduce the probability of an accident because the testing itself can increase the probability of an accident. It may also reduce accident consequences by increasing the equipment availability (i.e., less time in test). Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR1 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The use of an actual actuation signal to satisfy a SR has either no impact on, or increases the margin of plant safety by:

- a) Increasing mitigation equipment availability, and,
- b) Improving mitigation equipment reliability by potentially reducing wear caused by unnecessary testing.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "TR1" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR2
10 CFR 50.92 EVALUATION
FOR
RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change in accordance with NUREG-1431, Rev. 1, removes the requirement for a special report to be generated and submitted to the NRC. Reporting to the NRC will be done commensurate with the reporting requirements of 10CFR 50.72 and 50.73.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change is purely an administrative reporting change and cannot affect any accident probability or consequences.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

This change is purely an administrative reporting change and cannot create any new accident or affect any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change is purely an administrative reporting change and does not affect any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "TR2" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR3
10 CFR 50.92 EVALUATION
FOR
RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision is to remove reference to specific post-maintenance tests from the ITS. Post-maintenance testing programs are controlled via plant administrative procedures in accordance with licensee controlled document (ITS Section 5.4.1, Procedures) commitments to NRC RG 1.33, "Quality Assurance Program Requirements (Operation)" and ANS 3.2-ANSI N 18.7, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." Specific post-maintenance testing requirements are contingent on the type and scope of maintenance actually performed as well as the availability and viability of test equipment, techniques, etc. Removal of specific testing requirements from the ITS and reliance on normal post-maintenance testing programs addressed by licensee controlled documents allow flexibility to modify testing to address the circumstances of the maintenance performed while still assuring OPERABILITY of equipment returned to service,

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated;*
or
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated;*
or
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This is an administrative change which removes specific post-maintenance test requirements from TS. The testing, or equivalent testing, to assure equipment OPERABILITY prior to return to service would still be done as required by normal plant maintenance retest programs. Therefore, this change would not result in any increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

This is an administrative change and does not create a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change is an administrative change and does not affect any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "TR3" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers (1 Page)

NUREG-1431 Specifications that are not applicable (1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
3.6.1	3.6-1
3.6.2	3.6-3
3.6.3	3.6-8
3.6.4	3.6-16
3.6.5	3.6-17
3.6.6	3.6-18
3.6.7	3.6-22
3.6.8	3.6-24
3.6.9	N/A
3.6.10	N/A
3.6.11	N/A
3.6.12	N/A
3.6.13	N/A
3.6.14	N/A
3.6.15	N/A
3.6.16	N/A
3.6.17	N/A
3.6.18	N/A
3.6.19	N/A

Methodology (2 Pages)

Industry Travelers Applicable to Section 3.6

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-17, Rev. 1	Incorporated	3.6-2	NRC approved.
TSTF-30, Rev. 1	Incorporated	3.6-4	Not applicable to Wolf Creek and Callaway.
TSTF-45, Rev. 1	Incorporated	3.6-5	NRC approved.
TSTF-46, Rev. 1	Incorporated	3.6-7	NRC approved.
TSTF-51	Not incorporated	N/A	Not NRC approved as of traveler cut-off date.
TSTF-52	Incorporated	3.6-1	
TSTF-145	Not incorporated	N/A	NRC approved as of traveler cut-off date.
WOG-91	Incorporated	3.6-11, 3.6-12	

NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

Specification #	Specification Title	Comments
3.6.4.B	Containment Pressure (Subatmospheric)	
3.6.5B	Containment Temperature (Ice Condenser)	
3.6.5C	Containment Temperature (Subatmospheric)	
3.6.6B	Containment Spray (no credit for iodine removal)	
3.6.6C	Containment Spray (Ice Condenser)	
3.6.6D	Containment Spray (Subatmospheric QS)	
3.6.6E	Containment Spray (Subatmospheric RS)	
3.6.9	Hydrogen Mixing System	
3.6.10	Hydrogen Ignition System	
3.6.11	Iodine Cleanup System	
3.6.13	Shield Building Air Cleanup System	
3.6.14	Air Return System	
3.6.15	Ice Bed	
3.6.16	Ice Condenser Doors	
3.6.17	Divider Barrier Integrity	
3.6.18	Containment Recirculation Drains	
3.6.19	Shield Building	

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment (~~Atmospheric, Subatmospheric, Ice Condenser, and Dual~~)

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.1.1 Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.</p> <p>The leakage rate acceptance criterion is < 1.0 L. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are < 0.6 L, for the Type B and Type C tests, and < 0.75 L, for the Type A test.</p> <p>the Containment Leakage Rate Testing Program.</p>	<p>NOTE SR 3.0.2 is not applicable</p> <p><u>3.6-1</u></p> <p>In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions the Containment Leakage Rate Testing Program</p> <p><u>3.6-1</u></p>
<p>SR 3.6.1.2 Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.</p> <p>NOT USED</p>	<p>In accordance with the Containment Tendon Surveillance Program</p> <p><u>B-PS</u></p>

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks (~~Atmospheric, Subatmospheric, Ice Condenser, and Dual~~)

LCO 3.6.2^e . ~~Two~~ containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. Entry and exit is permissible to perform repairs on the affected air lock components.
 2. Separate Condition entry is allowed for each air lock.
 3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	-----NOTES----- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. -----	<p style="text-align: center;"><u>B</u></p> <p style="text-align: right;">(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Verify the OPERABLE door is closed in the affected air lock.	1 hour
	<p><u>AND</u></p> A.2 Lock the OPERABLE door closed in the affected air lock.	24 hours
	<p><u>AND</u></p> A.3 -----NOTE----- Air lock doors in high radiation areas may be verified locked closed by administrative means. ----- Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days

(continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1. <u>AND</u> C.2 Verify a door is closed in the affected air lock. <u>AND</u> C.3 Restore air lock to OPERABLE status.	Immediately 1 hour 24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.2.1</p> <p style="text-align: center;">-----NOTES-----</p> <p>1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.</p> <p>2. Results shall be evaluated against acceptance criteria of applicable to SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions with the Containment Leakage Rate Testing Program.</p> <p style="text-align: center;">-----</p> <p>Perform required air lock leakage rate testing in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.</p> <p>The acceptance criteria for air lock testing are:</p> <p>a. Overall air lock leakage rate is $\leq [0.05 L_s]$ when tested at $> P_s$.</p> <p>b. For each door, leakage rate is $\leq [0.01 L_s]$ when tested at $> [\text{psig}]$. the Containment Leakage Rate Testing Program</p>	<p style="text-align: center;"><u>3.6-1</u></p> <p style="text-align: center;">----- NOTE ----- SR 3.0.2 is not applicable</p> <p>In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions the Containment Leakage Rate Testing Program</p> <p style="text-align: center;"><u>3.6-1</u></p>
<p>SR 3.6.2.2</p> <p style="text-align: center;">----- NOTE -----</p> <p>Only required to be performed upon entry or exit through the containment air lock.</p> <p style="text-align: center;">-----</p> <p>Verify only one door in the air lock can be opened at a time.</p>	<p style="text-align: center;"><u>3.6-2</u></p> <p><u>184 days</u> <u>24 months</u></p>

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves (~~Atmospheric, Subatmospheric, Ice Condenser, and Dual~~)

NOTE

Not applicable to Main Steam Safety Valves (MSSVs), Main Steam Isolation Valves (MSIVs), Feedwater Isolation Valves (MFIIVs), Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves, and Atmospheric Dump Valves (ADVs)

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

NOTES

1. Penetration flow path(s) ~~except for [42] inch purge valve flow paths~~ 3.6-17
~~except no more than two of three flow paths for containment purge supply and exhaust and containment vacuum/pressure relief paths at one time~~ may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----NOTE----- Only applicable to penetration flow paths with two containment isolation valves. ----- One or more penetration flow paths with one containment isolation valve inoperable except for a containment purge supply and exhaust valve or shield building bypass pressure/vacuum relief valve leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours (continued)
		B-PS

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. (continued)</p>	<p>A.2 -----NOTES----- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ----- Verify the affected penetration flow path is isolated.</p>	<p><u>3.6-11</u></p> <p>Once per 31 days for isolation devices outside containment</p> <p><u>AND</u></p> <p>Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment</p>
<p>B. -----NOTE----- Only applicable to penetration flow paths with two containment isolation valves. ----- One or more penetration flow paths with two containment isolation valves inoperable except for a containment purge supply and exhaust valve or shield building bypass pressure/vacuum relief valve leakage not within limit.</p>	<p>B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p>	<p>1 hour</p> <p><u>B-PS</u></p>

(continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. -----NOTE----- Only applicable to penetration flow paths with only one containment isolation valve and a closed system. ----- One or more penetration flow paths with one containment isolation valve inoperable.</p>	<p>C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p> <p>AND</p> <p>C.2 -----NOTES----- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ----- Verify the affected penetration flow path is isolated.</p>	<p>[4] 72 hours <u>3.6-4</u></p> <p><u>3.6-11</u></p> <p>Once per 31 days</p>
<p>D. Shield building bypass leakage not within limit.</p>	<p>D.1 Restore leakage within limit.</p>	<p>4 hours <u>B-PS</u></p>
<p>E D. One or more penetration flow paths with one or more containment purge supply and exhaust and vacuum/pressure relief valves not within purge valve leakage limits.</p>	<p>E D.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p> <p>AND</p>	<p>24 hours <u>B-PS</u> <u>ED</u></p> <p>(continued)</p>

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E D. (continued)</p>	<p>E D.2 -----NOTES----- 1 Isolation devices in high radiation areas may be verified by use of administrative means. 2 Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means ----- Verify the affected penetration flow path is isolated. AND E D.3 Perform SR 3.6.3.7 for the resilient seal purge of vacuum/pressure relief valves closed to comply with Required Action ED.1.</p>	<p><u>ED</u> <u>3.6-11</u> Once per 31 days for isolation devices outside containment <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment Once per 92 day <u>B-PS</u> <u>3.6-17</u> <u>ED</u></p>
<p>F E. Required Action and associated Completion Time not met.</p>	<p>F E.1 Be in MODE 3. AND F E.2 Be in MODE 5.</p>	<p>6 hours <u>ED</u> 36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.3.1 Not Used Verify each [42] inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition E of this LCO.</p>	<p>31 days <u>3.6-17</u></p>
<p>SR 3.6.3.2 Verify each 48 inch containment purge supply and exhaust and 12 inch vacuum/pressure relief valve is closed, except when the [8] inch containment purge these valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.</p>	<p>31 days <u>B-PS</u> <u>3.6-17</u></p>
<p>SR 3.6.3.3 -----NOTE----- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ----- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.</p>	<p>31 days <u>3.6.5</u></p>

(continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.3.4 -----NOTE----- Valves and blind flanges in high radiation areas may be verified by use of administrative means. -----</p> <p>Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.</p>	<p style="text-align: center;"><u>3.6-5</u></p> <p>Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days</p>
<p>SR 3.6.3.5 Verify the isolation time of each power operated and each automatic power operated containment isolation valve that is not locked, sealed or otherwise secured in position is within limits.</p>	<p>In accordance with the Inservice Testing Program or 92 days</p> <p style="text-align: center;"><u>3.6-7</u> <u>3.6-12</u> <u>B-PS</u></p>
<p>SR 3.6.3.6 NOT USED Cycle each weight or spring loaded check valve testable during operation through one complete cycle of full travel, and verify each check valve remains closed when the differential pressure in the direction of flow is < [1.2] psid and opens when the differential pressure in the direction of flow is > [1.2] psid and < [5.0] psid.</p>	<p>92 days</p> <p style="text-align: center;"><u>B-PS</u></p>

(continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.3.7</p> <p>NOTE This surveillance is not required when the penetration flow path is isolated by a leak tested blank flange.</p> <p>Perform leakage rate testing for containment purge supply and exhaust and vacuum/pressure relief valves with resilient seals in accordance with the Containment Leakage Rate Testing Program.</p>	<p><u>3.6-13</u></p> <p>184 days AND Within 92 days after opening the valve <u>3.6-17</u></p>
<p>SR 3.6.3.8</p> <p>Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.</p>	<p><u>18</u> months</p>
<p>SR 3.6.3.9</p> <p>NOT USED Cycle each weight or spring loaded check valve not testable during operation through one complete cycle of full travel, and verify each check valve remains closed when the differential pressure in the direction of flow is $\leq [1.2]$ psid and opens when the differential pressure in the direction of flow is $> [1.2]$ psid and $< [5.0]$ psid.</p>	<p>18 months</p> <p><u>B-PS</u></p>
<p>SR 3.6.3.10</p> <p>Verify each <u>12</u> inch containment purge valve vacuum/pressure relief valve is blocked to restrict the valve from opening $> 50^\circ$.</p>	<p><u>18</u> months</p> <p><u>B-PS</u></p>

(continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE	FREQUENCY
SR 3.6.3.11 Not Used Verify the combined leakage rate for all shield building bypass leakage paths is \leq [L_s] when pressurized to $>$ [psig].	NOTE SR 3.0.2 is not applicable In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions B-PS

3.6 CONTAINMENT SYSTEMS

3.6.4A Containment Pressure (Atmospheric, Dual, and Ice Condenser)

LCO 3.6.4A Containment pressure shall be \geq 110 psig and \leq 112 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 1/2 hours <u>3.6-8</u>
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4A.1 Verify containment pressure is within limits.	12' hours

Containment Air Temperature (~~Atmospheric~~
~~and Dual~~)
 3.6.5A

PS

3.6 CONTAINMENT SYSTEMS

3.6.5A Containment Air Temperature (~~Atmospheric and Dual~~)

PS

LCO 3.6.5A Containment average air temperature shall be $\leq 120^{\circ}\text{F}$.

B-PS

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.5A.1 Verify containment average air temperature is within limit.	24 hours

3.6 CONTAINMENT SYSTEMS

3.6.6A Containment Spray and Cooling Systems (~~Atmospheric and Dual~~)
(~~Credit taken for iodine removal by the Containment Spray System~~)

LCO 3.6.6A Two containment spray trains and two containment fan cooling unit (CFCU) trains with either

a. Four CFCUs, or

b. Three CFCUs, each of the three supplied from a different vital bus

shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 84 hours
C. One <u>required</u> containment cooling CFCU train inoperable such that a <u>minimum of two CFCUs remain OPERABLE.</u>	C.1 Restore <u>required</u> <u>containment-cooling CFCU</u> train to OPERABLE status.	7 days <u>AND</u> <u>3.6-14</u> 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. Two One required containment cooling trains spray train inoperable and one required CFCU train inoperable such that a minimum of two CFCUs remain OPERABLE.</p>	<p>D.1 Restore one required containment cooling spray train to OPERABLE status.</p> <p><u>OR</u>.</p> <p>D.2 Restore one CFCU train to OPERABLE status such that four CFCUs or three CFCUs, each supplied by a different vital bus, are OPERABLE.</p>	<p>72 hours</p> <p style="text-align: center;"><u>3.6-14</u></p> <p>72 hours</p>
<p>E. Required Action and associated Completion Time of Condition C or D not met.</p>	<p>E.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>E.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>
<p>F. Two containment spray trains inoperable.</p> <p><u>OR</u></p> <p>Any combination of three or more trains inoperable. One containment spray train inoperable and two CFCU trains inoperable such that one or less CFCUs remain OPERABLE.</p> <p><u>OR</u></p> <p>One or less CFCUs OPERABLE</p>	<p>F.1 Enter LCO 3.0.3.</p>	<p>Immediately</p> <p style="text-align: center;"><u>3.6-14</u></p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6A.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6A.2	Operate each required containment cooling train fan unit CFCU for \geq 15 minutes.	31 days <u>PS</u>
SR 3.6.6A.3	Verify each [required] containment cooling train component cooling water flow rate to each required CFCU is \geq 1650 gpm.	31 days <u>PS</u>
SR 3.6.6A.4	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6A.5	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months <u>B</u>
SR 3.6.6A.6	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	18 months <u>B</u>
SR 3.6.6A.7	Verify each [required] containment cooling train CFCU starts automatically on an actual or simulated actuation signal.	18 months <u>3.6-14</u> <u>B</u>

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE	FREQUENCY				
SR 3.6.6A.8 Verify each spray nozzle is unobstructed.	<table border="1"> <tr> <td>At first</td> </tr> <tr> <td>refueling</td> </tr> <tr> <td>AND</td> </tr> <tr> <td>10 years</td> </tr> </table> <p><u>.PS</u></p>	At first	refueling	AND	10 years
At first					
refueling					
AND					
10 years					
SR 3.6.6.9 Verify each CFCU starts on low speed	31 days <p><u>3.6-14</u></p>				

3.6 CONTAINMENT SYSTEMS

3.6.7 Spray Additive System (~~Atmospheric, Subatmospheric, Ice Condenser, and Dual~~)

LCO 3.6.7 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.7.1 Verify each spray additive manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE	FREQUENCY
SR 3.6.7.2 Verify spray additive tank solution volume is \geq [2568] gal 46.2% and \leq [4000] gal 91.9%.	184 days <u>3.6-10</u> <u>B-PS</u>
SR 3.6.7.3 Verify spray additive tank NaOH solution concentration is \geq 30% and \leq 32% by weight.	184 days <u>B-PS</u>
SR 3.6.7.4 Verify each spray additive automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months <u>B</u>
SR 3.6.7.5 Verify spray additive flow [rate] from each solution's flow path.	5 years <u>B-PS</u>

Hydrogen Recombiners (~~Atmospheric, Subatmospheric,
Ice Condenser, and Dual~~)
3.6.8

PS

3.6 CONTAINMENT SYSTEMS

3.6.8 Hydrogen Recombiners (~~Atmospheric, Subatmospheric, Ice
Condenser, and Dual~~) (~~if permanently installed~~)

PS

LCO 3.6.8 Two hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One hydrogen recombiner inoperable.	A.1 -----NOTE----- LCO 3.0.4 is not applicable. ----- Restore hydrogen recombiner to OPERABLE status.	30 days
B. Two hydrogen recombiners inoperable.	B.1 Verify by administrative means that the hydrogen control function is maintained. <u>AND</u> B.2 Restore one hydrogen recombiner to OPERABLE status.	1 hour <u>AND</u> Once per 12 hours thereafter 7 days <u><u>B</u></u>
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Perform a system functional test for each hydrogen recombiner.	18 months B
SR 3.6.8.2	Visually examine each hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	18 months B
SR 3.6.8.3	Perform a resistance to ground test for each heater phase.	18 months B

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes- Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

**Methodology For Mark-up of NUREG-1431 Specifications
(Continued)**

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES CONTENTS

Mark-up:

<u>BASES</u>	<u>PAGE</u>
3.6.1	B3.6-1
3.6.2	B3.6-5
3.6.3	B3.6-11
3.6.4	B3.6-27
3.6.5	B3.6-32
3.6.6	B3.6-35
3.6.7	B3.6-45
3.6.8	B3.6-50
3.6.9	N/A
3.6.10	N/A
3.6.11	N/A
3.6.12	N/A
3.6.13	N/A
3.6.14	N/A
3.6.15	N/A
3.6.16	N/A
3.6.17	N/A
3.6.18	N/A
3.6.19	N/A
Methodology	(1 Page)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment (Atmospheric)

BASES

BACKGROUND The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat with a reactor cavity pit projection, and a shallow hemispherical dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

~~For containments with ungrouted tendons, the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed utilizing a three way post tensioning system.~~

~~The exterior shell and concrete reactor building structure around the reactor vessel (crane wall and bio-shield wall) is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. The steel liner additionally provides support and anchorage for safety related piping and electrical raceway. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions. The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:~~

- a. All penetrations required to be closed during accident conditions are either:
 1. capable of being closed by an OPERABLE automatic containment isolation system, or
 2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves"
- b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";
- c. All equipment hatches are closed; and
- d. ~~The pressurized sealing mechanism associated with a penetration (e.g. welds, bellows or O-rings) is OPERABLE, except as provided in LCO 3.6.[-].~~

APPLICABLE
SAFETY
ANALYSIS

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break and a red

(Continued)

BASES

ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA or a fuel handling accident. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_3 : the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_3) resulting from the limiting DBA LOCA. The allowable leakage rate represented by L_3 forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_3 is assumed to be 0.10% of containment air weight per day in the safety analysis at $P_3 = 47$ psig (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement 10CFR50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0 L_3$, except prior to the first startup after performing a required 10 CFR 50, Appendix J, Containment Leakage Rate Testing Program leakage test. At this time, the combined Type B and C leakage must be $< 0.6 L_3$, and the overall Type A leakage must be $< 0.75 L_3$, applicable Containment Leakage Rate Testing Program leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and containment purge supply and exhaust, hydrogen purge, and containment pressure/vacuum relief valves with resilient seals (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding

(Continued)

BASES

~~these individual limits only~~ result in the containment being inoperable when the leakage results in exceeding the Overall acceptance criteria of Appendix J 1.0 L_a.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements as specified in the Containment Leakage Rate Testing Program which is consistent with Reg Guide 1.163, 1995, and the requirements of 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions. Failure to meet air lock and purge valve with resilient seal leakage limits specified in the Containment Leakage Rate Test Program LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, Containment Leakage Rate Testing Program leakage test is required to be $< 0.6 L_a$ for combined Type B and C leakage following an outage or shutdown that included Type B and C testing only, and $\leq 0.75 L_a$ for overall Type A leakage following an outage or shutdown that included Type A testing. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0 L_a$. At $\leq 1.0 L_a$, the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by Appendix J, as modified by approved exemptions 10CFR50 App J Option B. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

BASES

- REFERENCES
1. 10 CFR 50, Appendix J, ~~Option B~~.
 2. FSAR, Chapter 15.
 3. FSAR, Section 6.2.
 4. ~~Regulatory Guide 1.35, Revision [1]~~.
 4. ~~Regulatory Guide 1.163 (September 1995)~~
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks (~~Atmospheric, Subatmospheric, Ice Condenser, and Dual~~)

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

~~There are two containment airlocks. Each [the personnel] air lock is nominally a right circular cylinder, approximately 9 ft in diameter, with a door at each end. The emergency air lock is approximately 5 ft 9 in inside diameter with a 2 ft 6 in door at each end. The On both air locks, doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).~~

~~Each personnel air lock is provided with limit switches on both doors that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated.~~

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

APPLICABLE
SAFETY
ANALYSIS

~~In MODE 1, 2, 3, and 4, the DBAs that results in a release of radioactive material within containment are as a loss of coolant accident and a rod ejection accident (Ref. 2). In the analysis of each of these this accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as $L_a = 0.1\%$ of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure $P_a = 48.3-47.0$ psig following a DBA LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.~~

~~The containment air locks satisfy Criterion 3 of the NRC Policy Statement 10CFR50.36(c)(2)(ii).~~

LCO Each containment air lock forms part of the containment pressure boundary. As part of containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

(Continued)

BASES

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.34, "Containment Penetrations."

ACTIONS The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the ~~limit for the air lock~~ then the leakage must be evaluated for its effect on the overall containment leakage rate. Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment" if the overall containment leakage limits are exceeded.

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTION of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion

(Continued)

BASES

Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to

(Continued)

BASES

evaluate previous combined leakage rates using current air lock test results. The air lock operability leakage limit is 0.05 L, and is considered part of the type B and C leakage and therefore subject to the containment inoperability limit of >1.0 L, under LCO 3.6.1. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established within the Containment Leakage Rate Testing Program during initial air lock and containment OPERABILITY testing under 10CFR50, Appendix J, Option B. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is also required by Appendix J (Ref. 1), as modified by approved exemptions.

(Continued)

BASES (Continued)

SURVEILLANCE
REQUIREMENTS
(Continued)

~~Thus, SR 3.0.2 (which allows Frequency extensions) does not apply the
Containment Leakage Rate Testing Program~~

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of the ~~Containment Leakage Rate Testing Program~~, which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the overall ~~combined Type B and C~~ containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the

(Continued)

BASES (Continued)

SURVEILLANCE
REQUIREMENTS

containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only not normally challenged when the containment air lock door is opened used for entry and exit (procedures require strict adherence to single door opening). This test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently than every 184 days every 24 months. The 24 month Frequency is based on avoiding the loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. The 184 day 24 month Frequency is based and on engineering judgement and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel given that the interlock is not challenged during use of the airlock.

REFERENCES

1. 10 CFR 50, Appendix J, ~~Option B~~.
 2. FSAR, Section ~~3.8, 6.2, and 15~~.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves (~~Atmospheric, Subatmospheric, Ice Condenser, and Dual~~)

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (~~typically containment isolation valves (A listing of containment isolation valves is provided in Technical Requirements Manual Table 2.1.1.)~~) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the ~~containment purge supply and exhaust valves, Hydrogen Purge and containment pressure/vacuum relief isolation purge and exhaust valves receive an a Containment Ventilation Isolation (CVI) signal on a containment high radiation condition. In addition to these large valves, the containment gas and particulate radiation monitor penetrations also isolate upon receipt of a CVI signal.~~ As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

Shutdown Containment Purge System (48 inch purge valves)

The ~~Shutdown Containment Purge System~~ operates to supply outside air into the containment for ventilation and cooling or heating needed for ~~prolonged containment access following a shutdown and during refueling and~~ The system may also be used to reduce the concentration of noble gases within containment prior to and during

(Continued)

BASES

BACKGROUND
(Continued)

personnel access. The supply and exhaust lines each contain two isolation valves. Because of their large size, the 48 inch Containment Purge valves in some units are not qualified for automatic closure from their open position under DBA conditions. Therefore, the 48 inch Containment Purge supply and exhaust isolation valves are normally maintained closed must be blocked to prevent opening more than 80% in MODES 1, 2, 3, and 4 to ensure closure within 2 seconds under DBA conditions (in order to support the required containment ventilation isolation time) and to ensure that the containment boundary is maintained. These valves may be opened as necessary to:

a. Reduce noble gases within containment prior to and during personnel access, and

b. Mitigate the effects of controller leakage and other sources which may effect the habitability of the containment for personnel entry.

Operation in MODES 1, 2, 3, or 4 with the 48 inch purge valves or the 12-inch vacuum/pressure relief valves open providing a flow path is limited to no more than 200 hours per calendar year.

Hydrogen Purge System (12-4 inch purge valves)

The Hydrogen Purge System is a supplementary system for the internal electric hydrogen recombiners and operated for hydrogen dilution or external hydrogen recombiners in for the containment following a LOCA. Because the 12-4 inch Containment Hydrogen Purge supply and exhaust valves are remote manually operated not qualified for automatic closure from their open position under DBA conditions they are normally maintained closed with power removed in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained.

Minipurge System Containment Pressure/Vacuum Relief (12 inch purge discharge isolation valves)

The Minipurge System Containment Pressure/Vacuum Relief valves are operated as necessary to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access, and

b. Equalize containment internal and external pressures.

Since the 12 inch Containment Pressure/Vacuum Relief valves used in the Minipurge System are designed to meet the requirements for automatic containment isolation valves within 5 seconds if mechanical blocks are installed to prevent opening more than 50%, these valves may be opened as needed in MODES 1, 2, 3, and 4.

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are in ~~MODES 1, 2, 3, or 4~~ is a loss of coolant accident (LOCA) and a ~~rod ejection accident~~ (Ref. 1). In the analyses for each of these ~~this~~ accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including ~~containment purge the Containment Purge, Hydrogen Purge, and Containment Vacuum/Pressure Relief valves~~) are minimized. ~~The safety analyses assume that if the 48 inch Containment Purge supply and exhaust valves close within 2 seconds and the 12 inch Hydrogen Purge pressure/vacuum relief valves are closed close within 5 seconds after the DBA at event initiation, the safety analysis shows that offsite dose release will be less than 10CFR100 guidelines.~~

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSES
(continued)

~~The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_d . The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.~~

~~The LOCA offsite dose analysis assumes leakage from the containment at a maximum leak rate of 0.10 percent of the containment volume per day for the first 24 hours, and at 0.05 percent of the containment volume per day for the duration of the accident. The Containment Purge supply and exhaust and the Containment Pressure Relief penetration are the only flow paths explicitly addressed in the dose analysis since it provides a direct activity release path from the containment to the environment. It is assumed to isolate within 5 seconds of receiving a containment isolation signal.~~

~~The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the 48 inch Containment Purge supply and exhaust and the 18 1/2 inch containment purge Containment Pressure/Vacuum Relief valves. Two valves in series on each purge line provide assurance that both the supply and exhaust line's could flow paths can be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with diverse power sources, motor operated and are pneumatically operated spring closed, respectively valves that will fail closed on the loss of power or air. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.~~

~~The 48 inch Containment Purge supply and exhaust and 12 inch Hydrogen Purge Containment Pressure/Vacuum Relief valves may be unable to close in the environment following a LOCA. Therefore, each of the Containment Purge supply and exhaust and Hydrogen Purge Containment Vacuum/Pressure Relief valves is required to remain sealed closed may be opened to provide a flow path. The 48 inch Containment Purge supply and exhaust valves and/or 12-inch vacuum/pressure relief valves may be open no more than 200 hours per calendar year while in during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valves due to failure in the control circuit associated with each valve. Again, The Purge Additionally, only two of the three flow paths (containment purge supply and exhaust, and containment vacuum/pressure relief) may be open at one time. The system valve design is designed to preclude a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.~~

~~The containment isolation valves satisfy Criterion 3 of the NRC Policy Statement 10CFR50.36(c)(2)(ii).~~

(Continued)

BASES (Continued)

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA. The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 48 inch Containment Purge supply and exhaust and 12 inch Hydrogen Purge valves and the Pressure/Vacuum Relief valves must be maintained sealed closed or have blocks installed to prevent full opening. These blocked purge valves also actuate on an automatic isolation signal. The valves covered by this LCO are listed along with their associated stroke times in the FSAR Technical Requirements Manual Plant Procedure AD13 DC1 Attachment 7.10 (Ref. 2 5).

The Normally closed passive containment isolation valves/devices are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 1.

Containment Purge supply and exhaust valves, Hydrogen Purge, and Containment Pressure/Vacuum Relief valves with resilient seals (and secondary containment bypass valves) must meet additional leakage rate surveillance frequency requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves and the Containment Purge supply and exhaust, Hydrogen Purge, and Containment Pressure/Vacuum Relief purge valves will perform their designed safety function to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

The LCO is modified by a Note stating that the Main Steam Safety Valves, Main Steam Isolation Valves, Feedwater Isolation Valves, and Atmospheric Dump Valves are not addressed in this LCO. These penetration flow paths credit the steam generators and piping inside containment as a containment isolation barrier (i.e., closed system). These valves are addressed by LCO 3.7.1 "Main Steam Safety Valves (MSSVs)", LCO 3.7.2 "Main Steam Isolation Valves (MSIVs)", LCO 3.7.3 "Main Feedwater Isolation Valves (MFIVs)", Main Feedwater Regulating Valves (MFRVs), and Associated Bypass Valves, and LCO 3.7.4 "Atmospheric Dump Valves (ADVs)" which provide the appropriate Required Actions in the event these valves are inoperable.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES.

(Continued)

APPLICABILITY (continued) Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS The ACTIONS are modified by a Note allowing penetration flow paths that are normally isolated by locked or sealed closed valves or valves that do not receive a containment isolation signal, except for 48-inch Containment Purge and 12-inch Hydrogen Purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that the 48-inch Containment Purge and the 12-inch Hydrogen Purge supply and exhaust valves are not qualified for automatic closure from their open position under DBA conditions and that these penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. A single valve in either the 48-inch Containment Purge or the 12-inch Hydrogen Purge a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1. This Note also limits operation of the normally isolated Containment Supply and Exhaust valves (2 penetration flow paths) and the Vacuum/Pressure Relief valves (1 penetration flow path) no more than 2 of 3 penetration flow paths open at one time.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event the air lock containment isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

(Continued)

BASES (Continued)

In the event one containment isolation valve in one or more penetration flow paths requiring isolation following a DBA is inoperable [~~except for purge Containment Purge supply and exhaust, Hydrogen Purge and Containment Pressure/Vacuum Relief isolation valve or shield building bypass leakage not within limit~~], the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve (this includes power operated valves with power removed), a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, ~~through a system walkdown,~~ that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

(Continued)

BASES (Continued)

A second Note has been added to Required Action A.2 to provide clarification that the action to periodically verify the affected penetration flow path is isolated does not apply to manual valves and blind flanges that are locked, sealed, or otherwise secured. This is acceptable since these were verified to be in the correct position prior to locking, sealing, or securing.

ACTIONS
(continued)

B.1

With two containment isolation valves in one or more penetration flow paths requiring isolation following a DBA inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve (this includes power operated valves with power removed), and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths requiring isolation following a DBA with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve (this includes power operated valves with power removed), and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the [4] 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4 (See FSAR Table 6.2-39 GDC-57 valves). In the event the affected penetration flow path

(Continued)

BASES (Continued)

is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. ~~The closed system must meet the requirements of Reference 3.~~ This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

ACTIONS
(continued)

Required Action C.2 is modified by a Note that applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

~~A second Note has been added to Required Action C.2 to provide clarification that the action to periodically verify the affected penetration flow path is isolated does not apply to manual valves and blind flanges that are locked, sealed, or otherwise secured. This is acceptable since these were verified to be in the correct position prior to locking, sealing, or securing.~~

D-1

~~With the shield building bypass leakage rate not within limit, the assumptions of the safety analyses are not met. Therefore, the leakage must be restored to within limit caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed to be the lesser actual pathway leakage of the two devices. The 4 hour Completion Time is reasonable considering the time required to restore the leakage by isolating the penetration(s) and the relative importance of secondary containment bypass leakage to the overall containment function.~~

ED.1, ED.2, and ED.3

~~In the event one or more Containment Purge supply and exhaust, Hydrogen Purge, or Containment Pressure/Vacuum Relief isolation valves in one or more penetration flow paths are not within the purge valve leakage limits, purge valve leakage must be~~

(Continued)

BASES (Continued)

restored reduced to within limits, or the affected penetration flow path must be isolated. For this ACTION, the leakage limit is as specified under the Containment Leakage Rate Testing Program and exceeding this limit would require evaluation per Note 4 under LCO 3.6.3. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a ~~FE~~ closed and de-activated automatic valve, closed manual valve (this includes power operated valves with power removed), or blind flange. A purge Containment Purge supply and exhaust, ~~Hydrogen Purge, or Containment Pressure/Vacuum Relief~~ Isolation valve with resilient seals utilized to satisfy Required Action ED.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.7. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.

ACTIONS
(continued)

In accordance with Required Action ED.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated leak rate following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur will not exceed the limit assumed in the offsite dose analysis. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the ~~Containment Purge supply and exhaust, Hydrogen Purge, or Containment Pressure/Vacuum Relief~~ Isolation valve with resilient seal that is isolated in accordance with Required Action ED.1, SR 3.6.3.7 must be performed at least once every 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase beyond the limits during the time the penetration is isolated. The normal Frequency for SR 3.6.3.7, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 3 4). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

FE.1 and FE.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5

(Continued)

BASES (Continued)

within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(Continued)

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.1 Not Used

~~Each 18 inch Containment Purge and 12 inch Hydrogen Purge valve is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a Containment Purge or Hydrogen Purge valve. Detailed analysis of the purge these valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A Containment Purge or Hydrogen Purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The Frequency is a result of an NRC initiative, Generic Issue B 24 (Ref. 4 5), related to containment purge valve use during plant operations. In the event Containment Purge or Hydrogen Purge valve leakage requires entry into Condition E, the Surveillance permits opening one purge valve in a penetration flow path to perform repairs.~~

SR 3.6.3.2

This SR ensures that the minipurge 18 inch Containment Pressure Relief 48 inch Containment Purge supply and exhaust and the 12 inch Containment Pressure/Vacuum Relief valves are closed as required or, if open, open for an allowable reason. If a purge or pressure relief valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the minipurge Containment Purge supply and exhaust or Containment Pressure Relief valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The minipurge Containment Purge supply and exhaust or Containment Pressure/Vacuum Relief valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure

(Continued)

BASES

that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, ~~through a system walkdown,~~ that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment ~~and not locked sealed, or otherwise secured~~ and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

~~Note 2 modifies the requirement to verify the blind flange on the fuel transfer canal. The refueling cavity areas in containment are flooded only during plant shutdown for refueling. The flange is only removed to support refueling operations and replaced after drainage of the canal when no more fuel transfers between the fuel handling building and the containment will occur. Once replaced, the flange is not removed again until the next refueling. Since the removal of this flange is limited to refueling operations and access to it is restricted during MODES 1, 2, 3, and 4, the probability~~

(Continued)

BASES

~~of it being mispositioned between refuelings is small. Therefore, it is reasonable that it is only required to be verified closed after each drainage of the canal.~~

SR 3.6.3.5

Verifying that the isolation time of each power operated and automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. ~~[- The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program. or 92 days.]~~

SR 3.6.3.6 Not Used

SURVEILLANCE
REQUIREMENTS
(Continued)

~~In subatmospheric containments, the check valves that serve a containment isolation function are weight or spring loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.6 requires verification of the operation of the check valves that are testable during unit operation. The Frequency of 92 days is consistent with the Inservice Testing Program requirement for valve testing on a 92 day Frequency.~~

SR 3.6.3.7

~~For Containment Purge supply and exhaust, Hydrogen Purge, and Containment Pressure/Vacuum Relief valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this these penetrations leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 3 4).~~

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

~~The leak rate acceptance criteria for the containment purge supply and exhaust, hydrogen purge, and containment pressure/vacuum relief valves are in accordance with the Containment Leakage Rate Testing Program.~~

(Continued)

BASES

SR 3.6.3.8

Automatic containment isolation valves close on a ~~Phase A, Phase B, or CVI~~ signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The ~~18~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the ~~18~~ month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.9 ~~Not Used~~

~~In subatmospheric containments, the check valves that serve a containment isolation function are weight or spring loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.9 verifies the operation of the check valves that are not testable during unit operation. The Frequency of 18 months is based on such factors as the inaccessibility of these valves, the fact that the unit must be shut down to perform the tests, and the successful results of the tests on an 18 month basis during past unit operation.~~

SR 3.6.3.10

~~Reviewer's Note: This SR is only required for those units with resilient seal purge valves allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices on the valves that are not permanently installed.~~

Verifying that each ~~[42]~~ ~~12~~ inch containment purge ~~pressure/vacuum relief~~ valve is blocked to restrict opening to ~~≤ [50]%~~ ~~50%~~ is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the ~~purge containment pressure/vacuum relief~~ valves must close to maintain containment leakage within the values assumed in the accident analysis. ~~At other times when purge valves are required to be capable of closing (e.g., during movement of recently irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The 18 month Frequency is appropriate because the blocking devices are not typically removed only during a refueling outage except during maintenance.~~

(Continued)

SR 3.6.3.11 Not Used

~~This SR ensures that the combined leakage rate of all shield building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix J). The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions (and therefore, the Frequency extensions of SR 3.0.2 may not be applied), since the testing is an Appendix J, Type C test. This SR simply imposes additional acceptance criteria.~~

~~[By pass leakage is considered part of L₁. [Reviewer's Note: Unless specifically exempted].]~~

REFERENCES

1. FSAR, Section 15.
 2. FSAR, Section 6.2.
 - ~~3. Standard Review Plan 6.2.4~~
 - ~~3 4. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."~~
 - ~~4. Generic Issue B 24, "Containment Purge Valve Reliability"~~
 - ~~5. Technical Requirements Manual 2-1 "Containment Isolation Valves - Diablo Canyon Power Plant Administrative Procedure AD13 DC1 Attachment 7.10"~~
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4A Containment Pressure (Atmospheric, Dual, and Ice Condenser)

BASES

BACKGROUND The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer modeled pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB (SLB at 30% power) generates the greatest mass and energy release rate. Thus, the LOCA SLB event bounds the SLB LOCA event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 17.716 psia (3.013 psig). This resulted in a maximum peak pressure from a LOCA of 53.9 SLB of 42.25 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a , results from the limiting LOCA SLB at 30% power. The maximum containment pressure resulting from the worst case LOCA, 44.1 SLB, 42.25 psig, does not exceed the containment design pressure, 55.47 psig.

The containment was also designed for an external pressure load equivalent to 3.52.5 psig. The inadvertent actuation of the Containment Spray System was

(Continued)

BASES

analyzed to determine the resulting reduction in containment pressure (sudden cooling of -1.8 psid). The initial pressure condition used in this analysis was -0.3 -1.7 psig. EOC 3.6.4 limits the operation of containment to equal to or less than -1.0 psig. This resulted in a minimum pressure inside containment of -2.8 -2.0 psig, which is less than the design load.

(Continued)

BASES

APPLICABLE
SAFETY
ANALYSIS
(continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment pressure satisfies Criterion 2 of the NRC Policy Statement ~~10CFR50.36(c)(2)(ii)~~.

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within ~~1 1/4~~ hours. The Required Action is necessary to return operation to within the bounds of the containment analysis. ~~The 1 1/4 hour Completion Time is reasonable to return pressure to normal. Consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.~~

B.1 and B.2

(Continued)

BASES

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within

(Continued)

BASES

ACTIONS
(continued)

6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.4A.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. FSAR, Section 6.2.
 2. 10 CFR 50, Appendix K.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.5A Containment Air Temperature (Atmospheric and Dual)

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray and Cooling systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed resulting in higher to avoid exceeding peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE
SAFETY
ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of the Containment Spray System, Residual Heat Removal System, and Containment Cooling System being rendered inoperable. A spectrum of SLBs were analyzed assuming the worst single active failure. The failure to close of one Main Steam Isolation Valve (MSIV) is the worst case single active failure for the SLB which results in the highest containment air temperature.

The limiting DBA for the maximum peak containment air temperature is an SLB. The initial containment average air temperature assumed in the design basis analyses

(Continued)

BASES (Continued)

(Ref. 1) is 120°F. This resulted in a maximum containment air temperature of [340.9]°F. The design temperature is [320]°F.

The containment design temperature is 271°F. The containment structure was analyzed to withstand the maximum peak temperature for the limiting DBA LOCA to ensure that it can contain the release of radioactive materials resulting from the accident. The containment structure was not analyzed for SLBs since it is a less limiting event structurally which were not considered design basis for containment structural design.

The spectrum of SLBs cases are temperature limit is used to establish the environmental qualification operating envelope for inside containment. The analysis shows that the peak containment temperature is 326°F (experienced during the MSLB at 70 % power). The performance of required safety-related equipment including the containment structure itself, is evaluated against this operating envelope to ensure the equipment can perform its safety function (Ref. 2). The maximum peak containment air temperature was calculated to exceed the containment design temperature for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA SLB.

The temperature limit is also used in the depressurization Containment external pressure analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System (Ref. 1).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a LOCA. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded. Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement 10CFR50.36(c)(11).

LCO

During a DBA, with an initial Maintaining the containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured ensures that the initial containment temperature assumed in the DBA analysis will not be violated.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

(Continued)

BASES (Continued)

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.5A.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using four temperature measurements. The four temperature measurement locations are per-selected from:

a. TE-85 or TE-86, approximately 100 ft elevation between crane wall and containment wall.

b. TE-87 or TE-88, approximately 100 ft elevation between steam generators.

c. TE-89 or TE-90, approximately 140 ft elevation near equipment hatch or stairs at 270°, respectively.

d. TE-91 or TE-92, approximately 184 ft elevation on top of steam generator missile barriers away from steam generators.

~~taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere.~~ The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

1. FSAR, Section 6.2.
 2. 10 CFR 50.49.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.6A Containment Spray and Cooling Systems (~~Atmospheric and Dual~~) (~~Credit taken for iodine removal by the Containment Spray System~~)

BASES

BACKGROUND

The Containment Spray and Containment Cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "~~Containment Atmosphere Cleanup,~~" GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1), ~~or other documents that were appropriate at the time of licensing (identified on a unit specific basis).~~

The Containment Cooling System and Containment Spray System are ~~is an~~ Engineered Safety Feature (ESF) systems. They are ~~it is~~ designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and the Containment Cooling System provide ~~redundant~~ diverse methods to limit and maintain post accident conditions to less than the containment design values.

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, ~~containment spray pump suction is transferred from the RWST to the containment sumps is supplied by manual realignment of the residual heat removal (RHR) pumps after the RWST is empty.~~

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature, and to reduce fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the ~~residual heat~~

(Continued)

BASES (Continued)

BACKGROUND
(continued)

~~removal RHR heat exchangers and containment spray heat exchangers, coolers.~~ Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment atmospheric heat removal.

The Spray Additive System injects an NaOH solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water ~~minimizes the evolution~~ maximizes the retention of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically by a containment High-3 High pressure signal or manually. ~~If an "S" signal is present, the high- high pressure signal automatically actuation opens the containment spray pump discharge valves, starts the two containment spray pumps. If containment pressure continues to increase, a "P" signal (Containment Pressure Hi-3-Hi-) generates a pump signal, opens the containment spray pump discharge valves, opens the spray additive tank outlet valves, initiates a phase "B" isolation signal, and begins the injection phase. A manual actuation of the Containment Spray System will begin the same sequence and can be initiated by requires the operator to actuate two separate switches on action from the main control board to begin the same sequence. The injection phase of containment spray continues until an RWST empty Low-Low level alarm is received. The Low-Low empty level alarm for the RWST actuates valves to align the Containment Spray System pump suction with the containment sump and/or signals the operator to manually align secure the system to the recirculation mode. The Containment Spray System in the recirculation mode maintains an equilibrium temperature between the containment atmosphere and the recirculated sump water. Operation of the Containment Spray System in the recirculation mode is controlled by the operator in accordance with the emergency operating procedures. After re-alignment of the RHR system to the containment recirculation sump, the associated RHR spray header isolation valve may be opened to allow continued spray operation of one train of spray utilizing the RHR pump to supply flow~~

Containment Cooling System

Two trains of containment fan cooling, ~~each consisting of two CFCUs with one shared CFCU for a total of five, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. The five CFCUs are powered from three separate vital buses, with two CFCUs on each of two vital buses and the remaining CFCU from the third vital bus. Each train of two fan units CFCU is supplied with cooling water from a one of two separate trains of essential service water (ESW) loops of component cooling water (CCW). Air is drawn into the coolers through the fan and discharged to the annulus ring which supplies the steam generator compartments.~~

(Continued)

BASES (Continued)

pressurizer compartment, and instrument tunnel reactor coolant pumps, and outside the secondary shield in the lower areas of containment.

BACKGROUND
(continued)

During normal operation, all four fan units three CFCUS are operating. The fans are normally operated at high speed with ESW CCW supplied to the cooling coils. The Containment Cooling System, operating in conjunction with the Containment Ventilation and Air Conditioning systems, is CFCUS are designed to limit the ambient containment air temperature during normal unit operation to less than the limit specified in LCO 3.6.5, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

In post accident operation following an actuation signal, the Containment Cooling System fans CFCUS are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere. The temperature of the ESW CCW is an important factor in the heat removal capability of the fan units.

APPLICABLE
SAFETY
ANALYSES

The Containment Spray System and Containment Cooling System limits the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the main steam line break (MSLB). The LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one worst case single failure, containment spray train for LOCA and the failure to close of one MSIV for the SLB ESF bus, which are is the worst case single active failure for the respective DBAs. [Ref. 3] and results in one train of the Containment Spray System and Containment Cooling System being rendered inoperable. For the LOCA case, the worst single failure is the failure of one SSPS train, which results in only one CSP and two CFCUS available. For SLB case, the worst single failure is the failure of one MSIV to close with two CSP and three CFCUS operating.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 46-12 42.25 psig (experienced during an MSLB at 30% power) compared to an allowable 47 psig. The analysis shows that the peak containment temperature is 340-85 326°F (experienced during an MSLB at 70% power) and is compared to the environmental qualifications of plant equipment. Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a unit specific power level of 100 102% for the LOCA with one containment spray train and two CFCUS operating. The limiting case MSLB analyses and evaluations

(Continued)

BASES (Continued)

APPLICABLE
SAFETY
ANALYSIS
(continued)

are based upon a unit specific power level of 30% or 70% with two one containment spray trains and one containment cooling train three CFCUS operating, and failure of one MSIV to close. Initial (pre-accident) containment conditions of 120°F and 4-5.13 psig are assumed. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a - 2-9 2-0 1-80 psid containment pressure decrease and is associated with the based on a sudden cooling effect of 70°F in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4A.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment "P" signal (High-3 High-High) pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time of [60] seconds includes diesel generator (DG) startup (for loss of offsite power), block sequenced loading of equipment, containment spray pump startup, and spray line filling (Ref. 4 3).

~~Containment cooling train~~The CFCUS performance for post accident conditions is given in Reference 4. The result of the analysis is that each train (two CFCUS) combined with one train of containment spray can provide 100% of the required peak cooling capacity during the post accident condition. ~~The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 5 4.~~

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-3 High-High pressure setpoint to achieving full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time of [60] seconds, includes signal delay, DG startup (for loss of offsite power), and service water component cooling water pump startup times (Ref. 6).

The Containment Spray System and the Containment Cooling System satisfies Criterion 3 of the NRC Policy Statement (10CFR50.36(c)(2)(ii)).

(Continued)

BASES (Continued)

LCO

During a DBA LOCA, a minimum of one containment cooling train ~~two CFCUs~~ and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Refs. 7.4). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling CFCU trains consisting of ~~four CFCUs or three CFCUs~~ each supplied by a different vital bus must be OPERABLE. Therefore, in the event of an accident, at least one train in each system of containment spray and one train of CFCUs (two CFCUs) operate, assuming the worst case single active failure occurs. Each Containment Spray System train typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal and automatically transferring. Upon actuation of the RWST empty alarm, the suction flowpath must be capable of being manually transferred to the containment sump.

Each Containment Cooling System CFCU typically includes demisters, cooling coils, dampers, fans, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains CFCUs.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Cooling System are ~~is~~ not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are ~~is~~ adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and low probability of a DBA occurring during this period.

(Continued)

BASES (Continued)

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

ACTIONS
(continued)B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

~~With one of the required containment cooling trains inoperable, the inoperable required containment cooling train must be restored to OPERABLE status within 7 days. With one CFCU train inoperable such that a minimum of two CFCUs remain operable, restore the required CFCUs to OPERABLE status within 7 days. The components in this degraded condition provide the heat removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of DBA occurring during this period.~~

The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

D.1 and D.2

BASES (Continued)

~~With two required containment cooling trains inoperable, one of these required containment cooling trains must be restored to OPERABLE status within 72 hours. With one train of containment spray inoperable and one train of CFCUs inoperable such that a minimum of two CFCUs remain OPERABLE, restore one required train to OPERABLE status within 72 hours. The components remaining in OPERABLE status in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System, the iodine removal function of the Containment Spray System, and the low probability of DBA occurring during this period.~~

ACTIONS
(continued)

E.1 and E.2

If the Required Action and associated Completion Time of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

~~With two containment spray trains or any combination of three or more containment spray and cooling trains inoperable one containment spray train inoperable and two CFCU trains inoperable such that one or less CFCUs remain OPERABLE, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.~~

SURVEILLANCE
REQUIREMENTS

SR 3.6.6A.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, ~~through a system walkdown,~~ that those valves outside containment (only check valves are inside containment) and capable of potentially being mispositioned are in the correct position.

(Continued)

BASES (Continued)

SR 3.6.6A.2

Operating each ~~required~~ containment cooling train fan unit CFCU for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the ~~containment cooling train~~ CFCUS occurring between surveillances. It has also been shown to be acceptable through operating experience.

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.6A.3

Verifying that each required ~~containment cooling train~~ ESW cooling flow rate to each cooling unit is CFCU is receiving the required component cooling water flow of \geq ~~700~~ 1650 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3 4). The component cooling water (CCW) system is hydraulically balanced during normal operation to ensure that at least 1650 gpm is delivered to each CFCU during a design bases event (DBA). The hydraulic system balance considers normal system alignments and the potential for any single active failure.

Operation of the CFCUS is permitted with lower CCW flow to the CFCUS during ASME Section XI testing or decay heat removal in MODE 4 with the residual heat removal heat exchangers in service. To support this conclusion a calculation was performed to evaluate containment heat removal with one train of containment spray OPERABLE and reduced CCW flow to three CFCUS. The calculation concluded that this configuration would provide adequate heat removal to ensure that the maximum design pressure of containment was not exceeded during a DBA in MODE 1. This analysis also determined that a single failure could not be tolerated during this condition and still assure that the maximum design pressure of containment would not be exceeded (Ref. 6).

The Frequency was developed considering the known reliability of the Cooling Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6A.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are

(Continued)

BASES (Continued)

normal tests of centrifugal pump performance required by Section XI Part 6 of the ASME O&M Code (Ref. 8 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.6A.5 and SR 3.6.6A.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment ~~"S"~~ ~~(High 1) 3 high-high~~ pressure signal with a coincident ~~"S"~~ signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

~~The surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance may be used to satisfy both requirements.~~

SURVEILLANCE
 REQUIREMENTS
 (continued)

SR 3.6.6A.7

This SR requires verification that each ~~[required] containment cooling train~~ ~~(FCU)~~ actuates upon receipt of an actual or simulated safety injection signal. The [18] month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.6A.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at ~~[the first refueling and at]~~ 10 year intervals is considered adequate to detect obstruction of the nozzles.

SR 3.6.6.9

(Continued)

BASES (Continued)

The CFCUs are designed to start or restart in low speed upon receipt of an SI signal. This limits the starting current and resultant load on the Emergency Diesel Generators. This allow optimum load sequencing of ESF equipment immediately following a DBA. This SR ensures that this feature is functioning properly. The 31 day frequency is selected based upon the normal operation of the CFCUs in high speed during power operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
 2. 10 CFR 50, Appendix K.
 3. FSAR, Section 6.2.1.
 4. FSAR, Section 6.2.2.
 5. ~~FSAR, Section [].~~
 6. ~~FSAR, Section [].~~
 7. ~~FSAR, Section [].~~
 8. ~~5 ASME, Boiler and Pressure Vessel Code, Section XI Operations and Maintenance Code, 1987 with OMA-1988 addenda, Part 6.~~
 6. ~~License Amendment 89 to DPR-80 and License Amendment 88 to DPR-82, 3/2/94.~~
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System (~~Atmospheric, Subatmospheric, Ice Condenser, and Dual~~)

BASES

BACKGROUND The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA ~~LOCA~~).

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray droplets from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the preferred spray additive. The NaOH added to the spray also ensures a pH value of between ~~8.0~~ 8.5 and ~~10-11.0~~ 10.0 of the solution recirculated from the containment sump. This pH band minimizes the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

Eductor Feed Systems Only

The Spray Additive System consists of one spray additive tank that is shared by the two trains of spray additive equipment. Each train of equipment provides a flow path from the spray additive tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line. ~~The eductors are designed to ensure that the pH of the spray mixture is between 8.5 and 11.0.~~

BACKGROUND
(continued)

Gravity Feed Systems Only

~~The Spray Additive System consists of one spray additive tank, two parallel redundant motor operated valves in the line between the additive tank and the refueling water storage tank (RWST), instrumentation, and recirculation pumps. The NaOH solution is added to the spray water by a balanced gravity feed from the additive tank through the connecting piping into a weir within the RWST. There, it mixes with the borated water flowing to the spray pump suction. Because of the hydrostatic balance between the two tanks, the flow rate of the NaOH is controlled by the volume per foot of height ratio of the two tanks. This ensures a spray mixture pH that is ≥ 8.5 and ≤ 11.0 .~~

(Continued)

BASES

The Containment Spray System actuation signal opens the valves from the spray additive tank to the spray pump suction or the containment spray pump start a manual containment spray initiation signal also opens the valves from the spray additive tank after a 5 minute delay. The 28 30% to 31 32% NaOH by weight solution is drawn into the spray pump eductor suction which inject it into the spray pump suction. The spray additive tank capacity provides for the addition of NaOH solution to all of the water sprayed from the RWST into containment. The percent solution and volume of solution sprayed into containment ensures a long term containment sump pH of ≥ 9.0 8.0 and ≤ 9.5 10.0. This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase of spray operation and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.

APPLICABLE
SAFETY
ANALYSES

The Spray Additive System is essential to the removal of airborne iodine within containment following a DBA LOCA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that 100% 56.3 a minimum 83% of the containment free volume is covered by the spray (Ref. 1).

The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.6. "Containment Spray and Cooling Systems."

The DBA analyses assume that one train of the Containment Spray System/Spray Additive tank volume solution is added to the remaining Containment Spray System flow path to achieve the minimum required containment recirculation sump solution pH of 8.0 prior to reaching RWST low-low level.

The Spray Additive System satisfies Criterion 3 of the NRC Policy Statement 10CFR50.36(c)(2)(ii).

LCO

The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA LOCA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow until the Containment Spray System suction path is switched from the RWST to the containment sump, and to raise the average long term containment sump spray solution pH to a level conducive to iodine removal retention in the liquid phase, namely, to between 8.5 8.0 and 11.0 10.5 10.0. This pH range maximizes the effectiveness of the iodine removal mechanism (from the containment atmosphere) without introducing

(Continued)

BASES

conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY In MODES 1, 2, 3, and 4, a DBA ~~LOCA~~ could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the ~~containment atmosphere~~ iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides

(Continued)

BASES

assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, ~~through a system walkdown~~, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. ~~The required volume may be surveilled using an indicated level band of 50 to 88% for the Spray Additive Tank which corresponds to the LCO 3.6.7 minimum and maximum limits adjusted conservatively for instrument accuracy of $\pm 0.3\%$. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and equipped with a low level alarm in the control room, so that there is high confidence that a substantial change in level below an acceptable value would be detected.~~

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(Continued)

BASES (Continued)

SR 3.6.7.5

To ensure that the correct operation of the Spray Additive System pH level is established in the borated water solution provided by the Containment Spray System, the flow rate in to the Spray Additive System eductors is verified once every 5 years by verifying that the solution flow path is not blocked from the RWST through test valve 8993 for each of the two flow paths. Flow of between 50 and 100 gpm through the eductor test loops (supplied from the RWST) simulates flow from the Chemical Additive Tank. The flow rate through the eductors is not critical because the entire Chemical Additive Tank contents is injected prior to isolation. This SR provides assurance that the correct amount of NaOH will be metered into the flow path upon Containment Spray System initiation. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow rate.

REFERENCES 1. FSAR, Chapter 6.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Recombiners (~~Atmospheric, Subatmospheric, Ice Condenser, and Dual~~) (~~if permanently installed~~)

BASES

BACKGROUND . The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen-oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombiner systems are provided. Each consists of controls located in the control room, a power supply and a recombiner. Recombination is accomplished by heating a hydrogen-air mixture above 1150°F. ~~The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner.~~ A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.1 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA. Hydrogen may accumulate in containment following a LOCA as a result of:

- a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;
- b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;
- c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

(Continued)

BASES (Continued)

- d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 3.5 v/o about 616 days after the LOCA and 4.0 v/o about 2 days later if no recombiner was functioning (Ref. 3). Initiating the hydrogen recombiners when the primary containment hydrogen concentration reaches 3.5 v/o will maintain the hydrogen concentration in the primary containment below flammability limits.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 4). The Hydrogen Purge System is similarly designed and constructed such that one of two redundant trains is Design Class I (for Quality and electrical power) but not redundant. As such, it is an adequate backup to the redundant hydrogen recombiners since it would be relied upon only in the event of a non-design basis condition.

The hydrogen recombiners satisfy Criterion 3 of the NRC Policy Statement 10CFR50.36(c)(2)(11).

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

(Continued)

BASES

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

B.1 and B.2

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by the containment Hydrogen Purge System. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. In addition, the alternate hydrogen control system capability must be verified once per 12 hours thereafter to ensure its continued availability. Both the initial verification and all subsequent verifications may be performed as an administrative check by examining logs or other information to determine the availability of the ~~key locked~~ alternate hydrogen control system. It does not mean to perform the surveillances are needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

(Continued)

BASES (Continued)

C.1

If the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.8.1

Performance of a system functional test for each hydrogen recombinder ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to $\geq 700^{\circ}\text{F}$ in ≤ 90 minutes. After reaching 700°F , the power is increased to maximum power for approximately 2 minutes and power is verified to be ≥ 60 kW.

Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.8.2

This SR ensures there are no physical problems that could affect recombinder operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failure involves loss of power, blockage of the internal flow, missile impact, etc.

A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.8.3

This SR, which is performed following the functional test of SR 3.6.8.1 requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is $\geq 10,000$ ohms.

The 18 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

(Continued)

BASES

REFERENCES

1. 10 CFR 50.44.
 2. 10 CFR 50, Appendix A, GDC 41.
 3. ~~Regulatory Guide 1.7, Revision 2.~~
 4. FSAR Section 6.2.5.
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Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(3 Pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.6

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER

JUSTIFICATION

- 3.6-1 This change supports implementation of the 10 CFR 50, Appendix J, Option B, for performance based leakage rate testing Option B by referencing the Containment Leakage Rate Testing Program described in the Administrative Controls section. This change is consistent with the CTS and Traveler TSTF-52.
- 3.6-2 Consistent with Traveler TSTF-17, Rev. 1, this change would extend the testing frequency of containment air lock interlock mechanisms from 184 days to 24 months and delete the SR note per implementation of Appendix J, Option B. SR 3.6.2.2 would be revised to require testing of the air lock door interlocks at an interval of 24 months. Typically, the interlock is installed after each refueling outage, verified OPERABLE with this surveillance and not disturbed until the next refueling outage. If the need for maintenance arises when the interlock is required, the performance of the interlock surveillance would be required following the maintenance. In addition, when an air lock is opened during times the interlock is required, the operator first verifies that one door is completely shut and the door seals pressurized before attempting to open the other door. Therefore, the interlock is not challenged except during actual testing of the interlock. Consequently, it should be sufficient to ensure proper operation of the interlock by testing the interlock on a 24 month interval.
- 3.6-3 Consistent with the CTS, a Note has been added to clarify that the valves listed are not addressed in LCO 3.6.3. These valves utilize the steam generators and associated piping as a closed system inside of containment. These valves have separate LCOs that provide the appropriate Required Actions in the event these valves are inoperable.
- 3.6-4 Consistent with traveler TSTF-30, Rev. 1, this change takes credit for a closed system for isolating a failed containment isolation valve. The change would extend the Completion Time for a closed system flow path with an inoperable isolation valve to 72 hours. General Design Criteria 57 allows the use of a closed system in combination with a containment isolation valve to provide two containment barriers against the release of radioactive material following an accident. Currently, LCO 3.6.3 does not allow the use of a closed system to isolate a failed containment isolation valve even though the closed system is subjected to Type A containment leakage testing, is missile protected, and is seismic Category I piping. A closed system also typically has flow through it during normal operation such that any loss of integrity could be continually observed through a leakage detection system within containment and during routine system walkdowns for closed systems outside containment. As such, the use of a closed system is no different from isolating a failed containment isolation valve by use of a single valve as specified in Required ACTION A.1. Therefore, LCO 3.6.3, Required ACTION C.1 is revised to allow 72 hours to isolate a failed valve associated with a closed system. This 72 hour period provides the necessary time to perform repairs on a failed containment isolation valve when relying on an intact closed system. A Completion Time of 72 hours is considered appropriate given that certain valves may be located inside containment, the reliability of the closed system, and that 72 hours is typically provided for losing one train of redundancy throughout the NUREG-1431. If the closed system and associated containment isolation valve were both inoperable, the plant would be in LCO 3.0.3 since there is no specific Condition specified.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.6

CHANGE NUMBER

JUSTIFICATION

- 3.6-5 This change is in accordance with TSTF-45, Rev. 1 and revises SR 3.6.3.3 and SR 3.6.3.4 to specify that only containment isolation valves that are not locked, sealed, or otherwise secured are required to be verified closed. The position of the locked, sealed, or otherwise secured valves was verified before the valves were locked, sealed, or otherwise secured.
- 3.6-6 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.6-7 This change is in accordance with TSTF-46, Rev. 1 and revises SR 3.6.3.5 to delete the reference to verifying the isolation time of "each power operated" containment isolation valve and only require verification of each "automatic isolation valve." Valves credited as containment isolation valves which are power operated (i.e., can be remotely operated) that do not receive a containment isolation signal do not have as isolation time as assumed in the accident analyses since they require operator action. Therefore, deleting reference to power operated isolation valve time testing reduces the potential for misinterpreting the requirements of this SR while maintaining the assumptions of the accident analyses.
- 3.6-8 Revises the Completion Time for the restoration of containment pressure from 1 hour to [4] hours. The [4] hour Completion Time is consistent with the CTS. The [4] hours [] allows the adequate time to take all Required Actions in a controlled manner.
- 3.6-9 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.6-10 Replaces the chemical additive tank volume limits in gallons with a tank level limits in percent [].
- 3.6-11 A new Note is added to ITS 3.6.3, Condition A.2 [and C.2] in accordance with Traveler WOG-91. The additional Note applies to isolation devices that are locked, sealed or otherwise secured in position and allows these devices to be verified closed by use of administrative means. It is sufficient to assume that initial establishment of component status (e.g., isolation valves closed) was performed correctly. Subsequently, verification is intended to ensure the component has not been inadvertently repositioned. Given that the function of locking, sealing, or securing components is to ensure the same avoidance of inadvertent repositioning, the periodic reverification should only be a verification of the administrative control that ensures that the component remains in the required state. It would be inappropriate to remove the lock, seal, or other means of securing the component solely to perform an active verification of the required state.
- 3.6-12 Consistent with SR 3.6.3.8, which provides that actuation position testing is not required for valves locked, sealed, or otherwise secured in their required position under administrative control, this change would provide that isolation time testing is not required for automatic containment isolation valves that are locked, sealed, or otherwise secured in their required position under administrative control. This change is consistent with WOG-91.
- 3.6-13 A clarifying note is added to SR 3.6.3.7 that would allow that leakage rate testing for containment purge valves with resilient seals is not required when the penetration flow path is isolated by a leak tested blind flange.
- 3.6-14 This change would incorporate plant specific operability criteria for containment fan cooler units required to meet design functional requirements. These requirements are contained in the CTS.
- 3.6-15 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.6-16 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.6-17 The ACTIONS and SRs of ITS 3.6.3 are modified to reflect DCP current license bases allowance to open at one time any 2 of 3 the DBA qualified 48 inch purge supply and/or exhaust flow paths and 12 inch vacuum/pressure relief flow paths.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.6

CHANGE
NUMBER

JUSTIFICATION

3.6-18	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
3.6-19	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
3.6-20	Not Used.
3.6-21	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
3.6-22	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
3.6-23	Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(3 Pages)

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.6-1	This change supports implementation of 10 CFR 50, Appendix J, Option B, approved in November 1995. Testing is performed in accordance with the Containment Leakage Rate Testing Program as described in the Administrative Controls section. This change is consistent with CTS and is in accordance with Traveler TSTF-52.	Yes	Yes, in CTS	Yes	Yes, approved as Amendment 111
3.6-2	This change would extend the testing frequency of containment air lock interlock mechanisms from 184 days to 24 months. This change is in accordance with TSTF-17, Rev. 1.	Yes	Yes	Yes	Yes
3.6-3	Consistent with the CTS, a Note has been added to clarify that the valves listed are not addressed in LCO 3.6.3. These valves utilize the steam generators and associated piping as a closed system inside of containment. These valves also have separate LCOs that provide the appropriate Required Actions in the event these valves are inoperable.	Yes	Yes	Yes	Yes
3.6-4	This change would extend the Completion Time for a closed system flow path with an inoperable isolation valve to 72 hours. This change is in accordance with TSTF-30, Rev. 1.	Yes	Yes	No, WC does not have GDC 57 valves	No, Callaway does not have GDC 57 valves
3.6-5	This change would revise SR 3.6.3.3 and SR 3.6.3.4 to specify that only containment isolation valves that are not locked, sealed, or otherwise secured are required to be verified closed. This change is in accordance with TSTF-45, Rev. 1.	Yes	Yes	Yes	Yes
3.6-6	Consistent with the current CPSES TS, a note is added to SR 3.6.3.4 to clarify that the blind flange on the fuel transfer canal need not be verified closed except after each drainage of the canal.	No	Yes	No	No
3.6-7	This change would revise SR 3.6.3.5 to delete the reference to verifying the isolation time of "each power operated" containment isolation valve and only require verification of each "automatic isolation valve." This change is in accordance with TSTF-46, Rev. 1.	Yes	Yes	Yes	Yes

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.6-8	This change would revise the Completion Time for the restoration of containment pressure from 1 hour to [4] hours. The [4] hour Completion Time is consistent with the CTS.	Yes	Yes	No, CTS has 1 hour completion time.	No, CTS has 1 hour completion time.
3.6-9	These portions of the specification do not apply since a containment cooling system is not part of the CPSES plant design.	No	Yes	No	No
3.6-10	This change would replace the chemical additive tank volume limits in gallons with tank level limits in percent [].	Yes	Yes	No -CTS in gallons	No - Callaway does not have this system.
3.6-11	This change would provide that the Required Action to periodically verify the affected penetration flow path is isolated does not apply to manual valves and blind flanges that are locked, sealed, or otherwise secured since these were verified to be in the correct position prior to locking, sealing, or securing.	Yes	Yes	Yes	Yes
3.6-12	Consistent with SR 3.6.3.8 which provides that actuation position testing is not required for valves locked, sealed, or otherwise secured in their required position under administrative control, this change would provide that isolation time testing is not required for automatic containment isolation valves that are locked, sealed, or otherwise secured in their required position under administrative control.	Yes	Yes	Yes	Yes
3.6-13	A clarifying note is added to SR 3.6.3.7 that would allow that leakage rate testing for containment purge valves with resilient seals is not required when the penetration flow path is isolated by a leak tested blind flange.	Yes	Yes	Yes	Yes
3.6-14	This change would incorporate DCPD specific OPERABILITY criteria for CFCU required to meet design functional requirements.	Yes	No	No	No
3.6-15	SR 3.6.6.7 would be modified to reflect Callaway plant specific requirements for cooling water automatic functions as well as containment cooler functions	No	No	No	Yes

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.6-16	Consistent with the CTS, this specification has been modified to reflect the Callaway recirculation fluid pH control system.	No	No	No	Yes
3.6-17	The ACTIONS and SRs of ITS 3.6.3 are modified to reflect DCPD current licensing bases allowance to open at one time any 2 of 3 the DBA qualified 48 inch purge supply and/or exhaust valves and 12 inch vacuum/pressure relief valves.	Yes	No	No	No
3.6-18	Consistent with CTS 4.6.1.7.2.a, a new SR is being added to perform leakage rate testing for the 36 inch containment purge valves and associated blank flange once per 24 months and following each re-installation of the blank flange.	No, CTS does not contain 24 month requirement	No, CTS does not contain 24 month requirement.	Yes	Yes
3.6-19	Consistent with CTS 4.6.1.7.4, SR 3.6.3.7 is being revised to cover the leak rate testing of the mini-purge valves with resilient seals after opening for testing once each 184 days and within 92 days after opening the valve.	No, DCPD does not have Mini-purge system	No, not in CTS	Yes	Yes
3.6-20	Not Used	N/A	N/A	N/A	N/A
3.6-21	Consistent with Wolf Creek CTS 4.6.2.3.b, SR 3.6.6.3 is being deleted and included as part of SR 3.6.6.7. The current licensing basis only requires for the flow rate to be tested as part of the actuation test every 18 months not the 31 days frequency required in the ITS 3.6.6.3.	No	No	Yes	No
3.6-22	This change deletes the Note to ITS 3.6.3, Condition A and B. deleted.	No, retain ITS wording	No, retain ITS wording	Yes	Yes
3.6-23	This change deletes the Note to ITS 3.6.3, Condition C.	No, retain ITS wording	No, retain ITS wording	Yes	Yes

JLS Conversion to Improved Technical Specifications

Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 3/4.7 - Plant Systems

ITS 3.7 - Plant Systems



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.7

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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(10 Pages)
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CROSS-REFERENCE TABLE FOR 3/4.7
Sorted by Current TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.1.1	LCO		01-01-A	3.7.1	LCO		3.7-01
3.7.1.1	APP			3.7.1	APP		
3.7.1.1	ACTION	New *note	01-02-LS1	3.7.1	ACTION	NOTE	
3.7.1.1	ACTION	a. (new)	01-04-LS3	3.7.1	ACTION	A.2,B.1&2	3.7-01
3.7.1.1	ACTION	b.	01-05-M			Not used	
3.7.1.1	ACTION	New	01-04-LS3	3.7.1	ACTION	A.1&2	
3.7.1.1	ACTION	# note	01-04-LS3	3.7.1	ACTION	A.2 NOTE 1	3.7-01
3.7.1.1	ACTION	New	01-06-M	3.7.1	ACTION	B.1&2	
4.7.1.1	SR		01-07-A	3.7.1.1	SR		
4.7.1.1	SR	New** note	01-05-M	3.7.1.1	SR	NOTE	
3.7.1.1	TABLE	3.7-1	01-04-LS3	3.7.1	Table	3.7.1-1	3.7-01
3.7.1.1	TABLE	3.7-1	01-09- LS31	3.7.1	Table	3.7.1-1	3.7-01
4.7.1.1	TABLE	3.7- 1*note	01-04-LS3			Not used.	
4.7.1.1	TABLE	3.7-2 **note	01-07-A	3.7.1.1	SR		
4.7.1.1	TABLE	3.7-2		3.7.1.1	Table	3.7.1-2	3.7-27
4.7.1.1	TABLE	3.7- 2*note	01-10-LG	3.7.1	BASES		
4.7.1.1	TABLE	3.7-2	01-11-R	FSAR			
3.7.1.2	LCO		02-01-LG	3.7.5	LCO		
3.7.1.2	LCO	a.	02-01-LG	3.7.5	BASES		
3.7.1.2	LCO	b.	02-01-LG	3.7.5	BASES		
3.7.1.2	APP		02-04-M	3.7.5	APP		

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Current TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.1.2	APP	*note (new)	02-04-M	3.7.5	LCO	NOTE	
3.7.1.2	ACTION	New	02-02-LS5	3.7.5	ACTION	A	
3.7.1.2	ACTION	New	02-03-M	3.7.5	ACTION	A.1	
3.7.1.2	ACTION	a.	02-10- LS21	3.7.5	ACTION	B.1, C.1 and C.2	
3.7.1.2	ACTION	a.(new)	02-03-M	3.7.5	ACTION	B.1	
3.7.1.2	ACTION	b.	02-10- LS21	3.7.5	ACTION	C.2 & C.1	
3.7.1.2	ACTION	c.	02-05-A	3.7.5	ACTION	D.1	
3.7.1.2	ACTION	New	02-04-M	3.7.5	ACTION	E	
4.7.1.2.1	SR	a.(2)	02-07-M	3.7.5.1	SR		
4.7.1.2.1	SR	a.(3)	02-09-A	3.7.5.1	SR		
4.7.1.2.1	SR	b.	02-08-LS6	3.7.5.2	SR		3.7-29
4.7.1.2.1	SR	b. *note	02-14-M	3.7.5.2	SR	NOTE	
4.7.1.2.1	SR	c *note	02-14-M	3.7.5.3	SR	NOTE	
4.7.1.2.1	SR	c.	02-11-A	3.7.5.3	SR and 3.7.5.4		
4.7.1.2.1	SR	c.	02-12-TR1	3.7.5.3	SR		
4.7.1.2.1	SR	c.**note	02-04-M	3.7.5.3	SR	NOTE	
4.7.1.2.1	SR	New	02-11-A	3.7.5.4	SR		
4.7.1.2.1	SR	New	02-12-TR1	3.7.5.4	SR		
3.7.1.3	LCO		03-01-LG	3.7.6	LCO		3.7-10
3.7.1.3	APP		02-04-M	3.7.6	APP		
3.7.1.3	ACTION	a.	02-04-M	3.7.6	ACTION	A & C	3.7-10
3.7.1.3	ACTION	a.	02-10- LS21	3.7.5	ACTION	F&H	3.7-09

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Current TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.1.3	ACTION	b.	02-04-M	3.7.6	ACTION	B&C	3.7-10
3.7.1.3	ACTION	b.	02-10-LS21	3.7.5	ACTION	G&H	3.7-09
4.7.1.3.1	SR			3.7.6.1	SR		3.7-10
4.7.1.3.2	SR			3.7.6.2	SR		3.7-10
4.7.1.3.3	SR		02-13-A	3.7.5.6	SR		3.7-09
3.7.1.4	LCO			3.7.18	LCO		
3.7.1.4	APP			3.7.18	APP		
3.7.1.4	ACTION			3.7.18	ACTION		
4.7.1.4	SR		04-01-M	3.7.18.1	SR		
4.7.1.4	Table	4.7-1	04-02-LS8	3.7.18.1	SR		
4.7.1.4	Table	4.7-1	04-01-M	3.7.18.1	SR		
3.7.1.5	LCO		05-04-A	3.7.2	LCO		
3.7.1.5	APP		05-01-LS9	3.7.2	APP		
3.7.1.5	ACTION	MODE 1	05-02-LS11	3.7.2	ACTION	A&B	
3.7.1.5	ACTION	MODE 2/3	05-02-LS11	3.7.2	ACTION	C	
3.7.1.5	ACTION	*note	05-02-LS11	3.7.3	ACTION	C. NOTE	
3.7.1.5	ACTION	MODE 2/3	05-05-M	3.7.2	ACTION	C&D	
4.7.1.5	SR		05-06-TR1	3.7.2.1	SR		
4.7.1.5	SR (new)	**NOTE	05-07-LS23	3.7.2.1	SR	NOTE	
4.7.1.5	SR	new	05-08-A	3.7.2.2	SR New		3.7-56

CROSS-REFERENCE TABLE FOR CTS 3/4.7
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<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.1.6	LCO		06-01-LG	3.7.4	Bases		
3.7.1.6	LCO		06-09-A	3.7.4	LCO		
3.7.1.6	APP		02-04-M	3.7.4	APP		
3.7.1.6	ACTION	a. (new)	06-02-LS14	3.7.4	ACTION	A.1 NOTE	
3.7.1.6	ACTION	a.	02-10-LS21	3.7.4	ACTION	D	
3.7.1.6	ACTION	a.	02-04-M	3.7.4	ACTION	A.1&D	
3.7.1.6	ACTION	b.	02-10-LS21	3.7.4	ACTION	D	
3.7.1.6	ACTION	b.	02-04-M	3.7.4	ACTION	B	3.7-6, 3.7-5
3.7.1.6	ACTION	New	06-05-LS24	3.7.4	ACTION	C&D	3.7-5, 3.7-6
3.7.1.6	ACTION	New	02-10-LS21	3.7.4	ACTION	C&D	3.7-5, 3.7-6
3.7.1.6	ACTION	New	02-04-M	3.7.4	ACTION	C&D	3.7-5, 3.7-6
4.7.1.6	SR		06-01-LG	3.7.4	Bases		
4.7.1.6	SR	a.		3.7.4.3	SR		3.7-50
4.7.1.6	SR	b.	06-06-LG	3.7.4	Bases		
4.7.1.6	SR	c.	06-06-LG	3.7.4.1	SR & Bases		
4.7.1.6	SR	New	06-04-M	3.7.4.2	SR		
3.7.1.7	LCO		07-01-A	3.7.3	LCO		
3.7.1.7	APP		07-02-LS37	3.7.3	APP		
3.7.1.7	ACTION	(new) *note	07-03-LS15	3.7.3	ACTION	NOTE	

CROSS-REFERENCE TABLE FOR CTS 3/4.7
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<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.1.7	ACTION		07-06-LS33	3.7.3	ACTION	A,B&C	
3.7.1.7	ACTION	a.	07-04-LS16	3.7.3	ACTION	A,B,orC.1	
3.7.1.7	ACTION	b.	07-04-LS16	3.7.3	ACTION	A,B,orC.1	
3.7.1.7	ACTION	c.	07-04-LS16	3.7.3	ACTION	A,B,&C.1	
3.7.1.7	ACTION	(new)	07-04-LS16	3.7.3	ACTION	D	
3.7.1.7	ACTION	d.		3.7.3	ACTION	E.1&E.2	
3.7.1.7	ACTION	(new)	07-11-M	3.7.3	ACTION	A,B,orC.1	
4.7.1.7.1	SR		07-13-TR1	3.7.3.2	SR		3.7-3
4.7.1.7.2	SR		07-13-TR1	3.7.3.1	SR		
4.7.1.7.3	SR	(new)	05-08-A	3.7.3.3	SR New		3.7-56
3.7.3.1	LCO			3.7.7	LCO		
3.7.3.1	APP			3.7.7	APP		
3.7.3.1	ACTION	(new) *note	08-02-A	3.7.7	ACTION	A.1 NOTE	
3.7.3.1	ACTION			3.7.7	ACTION	A.1, B.1, and B.2	
4.7.3.1	SR	a. (new) note	08-04-A	3.7.7.1	SR	NOTE	
4.7.3.1	SR	a.	08-08-A	3.7.7.1	SR		
4.7.3.1	SR	b.	08-08-A	3.7.7.2	SR		
4.7.3.1	SR	b.	08-05-A	3.7.7.2	SR		
4.7.3.1	SR	b.	08-06-TR1	3.7.7.2	SR		
4.7.3.1	SR	(new)	08-07-M	3.7.7.3	SR		
3.7.4.1	LCO			3.7.8	LCO		3.7-30

CROSS-REFERENCE TABLE FOR CTS 3/4.7
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<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.4.1	APP			3.7.8	APP		
3.7.4.1	ACTION	(new) *note	08-02-A	3.7.8	ACTION	A.1 NOTE	3.7-13
3.7.4.1	ACTION			3.7.8	ACTION	A, B.1, B.2	
4.7.4.1	SR		09-03-A	3.7.8.1	SR		3.7-15
4.7.4.1	SR		09-01-M	3.7.8.1	SR		
4.7.4.1	SR	(new)	09-04-M	3.7.8.3	SR		
4.7.4.1	SR	(new)	09-06-M	3.7.8.2	SR		3.7-16
3.7.5.1	LCO		10-01-LG	3.7.10	LCO		
3.7.5.1	LCO	*note	10-13-LG	3.7.10	BASES		
3.7.5.1	LCO	**note	10-13-LG	3.7.10	BASES		
3.7.5.1	APP		10-02-M	3.7.10	APP		
3.7.5.1	ACTION	MODE 1-4		3.7.10	ACTION	A.1,B.1&2	
3.7.5.1	ACTION	New M. 1-4	10-04-A	3.7.10	ACTION	H	
3.7.5.1	ACTION	M. 5/6	10-02-M	3.7.10	ACTION	C	
3.7.5.1	ACTION	M. 5/6 a.	10-05-LS18	3.7.10	ACTION	A&C. 2.1 and 2.2	
3.7.5.1	ACTION	M. 5/6 b.	10-06-LG	3.7.10	BASES		
3.7.5.1	ACTION	M. 5/6 b.	10-02-M	3.7.10	ACTION	G	
3.7.5.1	ACTION	M. 5/6 b.	10-16-LG	3.7.10	BASES		
3.7.5.1	ACTION	M. 5/6b.	10-21-LS38	N/A		Not used.	
4.7.5.1	SR	a.	10-07-R	N/A		Not used.	
4.7.5.1	SR	b.1)	10-06-LG	3.7.10	BASES		
4.7.5.1	SR	b.1)		3.7.10.1	SR		

CROSS-REFERENCE TABLE FOR CTS 3/4.7
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<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
4.7.5.1	SR	b.2)	10-16-LG	3.7.10	BASES		
4.7.5.1	SR	b.3)	10-06-LG	3.7.10	BASES		
4.7.5.1	SR	b. (new)	10-08-A	3.7.10.2	SR		
4.7.5.1	SR	c. 1)	10-08-A	5.5.11			
4.7.5.1	SR	c. 2)	10-23-LS13	N/A		Not used.	
4.7.5.1	SR	c. 2)	10-24-M	5.5.11			
4.7.5.1	SR	c. 3)	10-17-A	5.5.11			
4.7.5.1	SR	d.	10-08-A	5.5.11			
4.7.5.1	SR	d.	10-23-LS13	N/A		Not used.	
4.7.5.1	SR	d.	10-24-M	5.5.11			
4.7.5.1	SR	e. 1)	10-08-A	5.5.11			
4.7.5.1	SR	e. 2)	10-10-TR1	3.7.10.3	SR		3.7-35
4.7.5.1	SR	e. 2)	10-08-A	3.7.10.3	SR		3.7-35
4.7.5.1	SR	e. 3)	10-11-LS19	3.7.10.4	SR		3.7-33
4.7.5.1	SR	e. 4)	10-08-A	5.5.11			
4.7.5.1	SR	f)	10-08-A	5.5.11			
4.7.5.1	SR	g)	10-08-A	5.5.11			
3.7.6.1	LCO		12-10-M	3.7.12	LCO		
3.7.6.1	APP			3.7.12	APP		
3.7.6.1	ACTION	a.		3.7.12	ACTION	A&C.1&2	3.7-21
3.7.6.1	ACTION	b.	12-10-M	3.7.12	ACTION	B&C.1&2	
4.7.6.1	SR	a. 1)	12-06-LG	3.7.12.1	SR		
4.7.6.1	SR	a. 2)	12-06-LG	3.7.12	Bases		

CROSS-REFERENCE TABLE FOR CTS 3/4.7
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<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
4.7.6.1	SR	a. (new)	10-08-A	3.7.12.2	SR		
4.7.6.1	SR	b. 1)	10-08-A	5.5.11			
4.7.6.1	SR	b. 2)	10-08-A	5.5.11			
4.7.6.1	SR	b. 2)	10-23-LS13	N/A		Not used.	
4.7.6.1	SR	b. 2)	10-24-A	5.5.11			
4.7.6.1	SR	b. 3)	10-08-A	5.5.11			
4.7.6.1	SR	b. 3)	10-17-A	5.5.11			
4.7.6.1	SR	c.	10-08-A	5.5.11			
4.7.6.1	SR	c.	10-23-LS13	N/A		Not used.	
4.7.6.1	SR	c.	10-24-A	5.5.11			
4.7.6.1	SR	d. 1)	10-08-A	5.5.11			
4.7.6.1	SR	d. 2)	12-10-M	5.5.11			
4.7.6.1	SR	d. 2)	12-04-TR1	3.7.12.3	SR		3.7-22
4.7.6.1	SR	d. 3)	10-08-A	5.5.11			
4.7.6.1	SR	d. 4)		3.7.12.6	SR new		3.7-18
4.7.6.1	SR	e.	10-08-A	5.5.11			
4.6.7.1	SR	f.	10-08-A	5.5.11			
3.7.12	LCO	*note	13-01-LG	3.7.9	LCO & Bases		
3.7.12	APP		13-08-M	3.7.9	APP		
3.7.12	ACTION		13-08-M	3.7.9	ACTION	A & B	3.7-17
3.7.12	ACTION		13-09-A			Not used.	
4.7.12	SR	a.		3.7.9.2	SR		3.7-17
4.7.12	SR	b.		3.7.9.2	SR		3.7-17
4.7.1.2	SR	c.		3.7.9.2	SR		3.7-17

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Current TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.9.12	LCO			3.7.13	LCO		
3.9.12	APP			3.7.13	APP		
3.9.12	ACTION	a.		3.7.13	ACTION	A	3.7-43
Not Used				3.7.13	ACTION	B Not Used	
Not Used				3.7.13	ACTION	C Not Used	3.7-43
3.9.12		b.		3.7.13	ACTION	G	
3.9.12	ACTION	c.		3.7.13	APP	NOTE	3.7-42
3.9.12	ACTION	(new)		3.7.13	ACTION (new)	D, E (NOT USED), & F	3.7-57
4.9.12	SR	a.		3.7.13.1	SR		
4.9.12	SR	(new)		3.7.13.2	SR		
4.9.12	SR	b.		5.5.11			
4.9.12	SR	c.		5.5.11			
4.9.12	SR	d. 1)		5.5.11			
4.9.12	SR	d. 2)		3.7.13.3	SR		
4.9.12	SR	d. 3)		3.7.13.4	SR		3.7-49
4.9.12	SR	e.		5.5.11			
4.9.12	SR	f.		5.5.11			
Not Used				3.7.13.5	SR	Not used.	
Not Used				3.7.14		Not used.	3.7-50
3.9.11	LCO			3.7.15	LCO		

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Current TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.9.11	APP			3.7.15	APP		
3.9.11	ACTION	a.		3.7.15	ACTION	A	
3.9.11	ACTION	b.		3.7.15	ACTION	NOTE	
4.9.11	SR			3.7.15.1	SR		
3.9.14.2	LCO			3.7.16	LCO		
3.9.14.2	APP			3.7.16	APP		3.7-53
3.9.14.2	ACTION	a.		3.7.16	ACTION	A	3.7-53
3.9.14.2	ACTION	b.		3.7.16	ACTION	NOTE	
4.9.14.2	SR			3.7.16.1	SR		3.7-53
3.9.14.1	LCO			3.7.17.2	LCO		3.7-54
3.9.14.1	APP			3.7.17.2	APP		
3.9.14.1	ACTION	a.		3.7.17.2	ACTION	A	3.7-54
3.9.14.1	ACTION	b.		3.7.17.2	ACTION	NOTE	
4.9.14.1	SR			3.7.17.2. 1	SR		
3.9-2	FIGURE			3.7.17-2	FIGURE		
3.9.14.3	LCO	a.		3.7.17.1	LCO	a.	3.7-51
3.9.14.3	LCO	b.		3.7.17.1	LCO	b.	3.7-51
3.9.14.3	APP			3.7.17.1	APP		3.7-51
3.9.14.3	ACTION	a.		3.7.17.1	ACTION	A	3.7-51
3.9.14.3	ACTION	b.		3.7.17.1	ACTION	NOTE	3.7-51
4.9.14.3	SR			3.7.17.1. 1	SR		3.7-51
3.9-1	FIGURE			3.7.17-1	FIGURE		3.7-51

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.1.1	LCO		01-01-A	3.7.1	LCO		3.7-01
3.7.1.1	APP			3.7.1	APP		
3.7.1.1	ACTION	New *note	01-02-L1	3.7.1	ACTION	NOTE	
3.7.1.1	ACTION	a.		3.7.1	ACTION	A.1	3.7-01
3.7.1.1	ACTION	New #note	01-04-LS3	3.7.1	ACTION	A.2 NOTE 1	3.7-01
3.7.1.1	ACTION	New ##note	01-04-LS3	3.7.1	ACTION	A.2 NOTE 2	3.7-01
3.7.1.1	ACTION	a. & New	01-04-LS3	3.7.1	ACTION	A.2	3.7-01
3.7.1.1	ACTION	a. & New	01-04-LS3	3.7.1	ACTION	B.1	
3.7.1.1	ACTION	a. & New	01-06-M	3.7.1	ACTION	B.2	
4.7.1.1	SR	** note	01-05-M	3.7.1.1	SR	NOTE	
4.7.1.1	SR		01-07-A	3.7.1.1	SR		
3.7.1.1	TABLE	3.7-1	01-04-LS3	3.7.1	Table	3.7.1-1	3.7-01
3.7.1.1	TABLE	3.7-1	01-09- LS31	3.7.1	Table	3.7.1-1	3.7-01
4.7.1.1	TABLE	3.7-2		3.7.1.1	Table	3.7.1-2	3.7-27
3.7.1.5	LCO			3.7.2	LCO		
3.7.1.5	APP		05-01-LS9	3.7.2	APP		
3.7.1.5	ACTION	MODE 1	05-02- LS11	3.7.2	ACTION	A	
3.7.1.5	ACTION	MODE 1		3.7.2	ACTION.	B	
3.7.1.5	ACTION	MODE 2/3	05-02- LS11	3.7.2	ACTION.	C. NOTE	
3.7.1.5	ACTION	MODE 2/3	05-05-M	3.7.2	ACTION	C.1	

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item.</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.1.5	ACTION	MODE 2/3	05-05-M	3.7.2	ACTION	C.2	
3.7.1.5	ACTION	MODE 2/3		3.7.2	ACTION	D.1	
3.7.1.5	ACTION	MODE 2/3		3.7.2	ACTION	D.2	
4.7.1.5	SR	**NOTE	05-07- LS23	3.7.2.1	SR	NOTE	
4.7.1.5	SR		05-06-TR1	3.7.2.1	SR		3.7-56
4.7.1.5	SR	New	05-08-A	3.7.2.2	SR	New	3.7-56
3.7.1.7	LCO		07-01-A	3.7.3	LCO		
3.7.1.7	APP		07-02-A	3.7.3	APP		
3.7.1.7	ACTION	New *note	07-03- LS15	3.7.3	ACTION	NOTE	
3.7.1.7	ACTION		07-06- LS33	3.7.3	ACTION.	A.1	
3.7.1.7	ACTION	New	07-11-M	3.7.3	ACTION	A.2	
3.7.1.7	ACTION			3.7.3	ACTION	B.1	
3.7.1.7	ACTION	New	07-11-M	3.7.3	ACTION.	B.2	
3.7.1.7	ACTION			3.7.3	ACTION	C.1	
3.7.1.7	ACTION	New	07-11-M	3.7.3	ACTION	C.2	
3.7.1.7	ACTION	d.	07-04- LS16	3.7.3	ACTION.	D	
3.7.1.7	ACTION	d.		3.7.3	ACTION	E.1	
3.7.1.7	ACTION	d.		3.7.3	ACTION	E.2	
4.7.1.7.2	SR		07-13-TR1	3.7.3.1	SR		3.7-3 & 3.7-56
4.7.1.7.1	SR		07-13-TR1	3.7.3.2	SR		3.7-3 & 3.7-56

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
4.7.1.7.1	SR	New	05-08-A	3.7.3.3	SR	New	3.7-56
3.7.1.6	LCO		06-09-A	3.7.4	LCO		
3.7.1.6	APP		02-04-M	3.7.4	APP		
3.7.1.6	ACTION	a.	06-02-LS14	3.7.4	ACTION	A.1 NOTE	
3.7.1.6	ACTION	a.	02-10-LS21, 02-04-M	3.7.4	ACTION	A	
3.7.1.6	ACTION	b.	02-10-LS21	3.7.4	ACTION	B	3.7-5
3.7.1.6	ACTION	b.	02-04-M	3.7.4	ACTION	B	3.7-6
3.7.1.6	ACTION	New	06-05-LS24	3.7.4	ACTION	C (New)	3.7-5
3.7.1.6	ACTION	New	06-05-LS24	3.7.4	ACTION	C (New)	3.7-6
3.7.1.6	ACTION	a.	02-10-LS21	3.7.4	ACTION	D.1	3.7-6
3.7.1.6	ACTION	b.	02-10-LS21	3.7.4	ACTION	D.2	3.7-6
4.7.1.6	SR	c.	06-06-LG	3.7.4.1	SR		
4.7.1.6	SR	New	06-04-M	3.7.4.2	SR		
4.7.1.6	SR	a.		3.7.4.3	SR	New	3.7-50
3.7.1.2	LCO		02-01-LG	3.7.5	LCO		
3.7.1.2	APP	*note	02-04-M	3.7.5	LCO	NOTE	
3.7.1.2	APP		02-04-M	3.7.5	APP		
3.7.1.2	ACTION	New	02-02-LS5	3.7.5	ACTION.	A	
3.7.1.2	ACTION	a.	02-03-M	3.7.5	ACTION	B	

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.1.2	ACTION	a. & b.	02-10-LS21	3.7.5	ACTION	C.1	
3.7.1.2	ACTION	a. & b.	02-10-LS21, 02-04-M	3.7.5	ACTION	C.2	
3.7.1.2	ACTION	c.	02-05-A	3.7.5	ACTION	D NOTE	
3.7.1.2	ACTION	c.		3.7.5	ACTION	D	
3.7.1.2	ACTION	New	02-04-M	3.7.5	ACTION.	E	
3.7.1.3	ACTION	a.	02-10-LS21, 02-04-M	3.7.5	ACTION	F	3.7-9
3.7.1.3	ACTION	b.	02-04-M	3.7.5	ACTION	G	3.7-9
3.7.1.3	ACTION	a. & b.	02-10-LS21, 02-04-M	3.7.5	ACTION	H.1	3.7-9
3.7.1.3	ACTION	a. & b.	02-10-LS21, 02-04-M	3.7.5	ACTION	H.2	3.7-9
4.7.1.2.1	SR	a.(2)	02-07-M, 02-09-A	3.7.5.1	SR		
4.7.1.2.1	SR	b. *note	02-14-M	3.7.5.2	SR	NOTE	
4.7.1.2.1	SR	b.	02-08-LS6	3.7.5.2	SR		3.7-29
4.7.1.2.1	SR	c. **note	02-04-M	3.7.5.3	SR	NOTE	
4.7.1.2.1	SR	c.	02-12-TR1, 02-11-A	3.7.5.3	SR		
4.7.1.2.1	SR	(new) *note	02-14-M	3.7.5.4	SR	NOTE 1	
4.7.1.2.1	SR	(new) **note	02-04-M	3.7.5.4	SR	NOTE 2	
4.7.1.2.1	SR	(new)	02-12-TR1, 02-11-A	3.7.5.4	SR		

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item.</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
Not Used				3.7.5.5	SR	NOT USED	
4.7.1.3.3	SR		02-13-A	3.7.5.6	SR		3.7-9
3.7.1.3	LCO		03-01-LG	3.7.6	LCO		3.7-10
3.7.1.3	APP		02-04-M	3.7.6	APP		
3.7.1.3	ACTION	a.	02-04-M	3.7.6	ACTION	A	3.7-10
3.7.1.3	ACTION	b.	02-04-M	3.7.6	ACTION	B	3.7-10
3.7.1.3	ACTION	b.	02-10-LS	3.7.5	ACTION	C.1	
3.7.1.3	ACTION	a. & b.	02-04-M	3.7.5	ACTION	C.2	
4.7.1.3.1	SR			3.7.6.1	SR		3.7-10
4.7.1.3.2	SR			3.7.6.2	SR		3.7-10
3.7.3.1	LCO			3.7.7	LCO		
3.7.3.1	APP			3.7.7	APP		
3.7.3.1	ACTION	New *note	08-02-A	3.7.7	ACTION	A.1 NOTE	
3.7.3.1	ACTION			3.7.7	ACTION	A	
3.7.3.1	ACTION			3.7.7	ACTION	B.1&2	
4.7.3.1	SR	New **note	08-04-A	3.7.7.1	SR	NOTE	
4.7.3.1	SR	a.	08-04-A	3.7.7.1	SR		
4.7.3.1	SR	b.	08-06- TR1, 08- 05-A, 08-08-A	3.7.7.2	SR		
4.7.3.1	SR	New	08-07-M	3.7.7.3	SR		

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.4.1	LCO			3.7.8	LCO		
3.7.4.1	APP			3.7.8	APP		
N/A				3.7.8	ACTION	A.1 NOTE 1	3.7-13
3.7.4.1	ACTION	New *note	08-02-A	3.7.8	ACTION	A.1 NOTE 2	
3.7.4.1	ACTION			3.7.8	ACTION	A	
3.7.4.1	ACTION			3.7.8	ACTION	B.1	
3.7.4.1	ACTION			3.7.8	ACTION	B.2	
N/A				3.7.8.1	SR	NOTE	3.7-14
4.7.4.1	SR		09-03-M	3.7.8.1	SR		3.7-15
4.7.4.1	SR	New	09-06-M	3.7.8.2	SR		3.7-16
4.7.4.1	SR	New	09-04-M	3.7.8.3	SR		
3.7.12	LCO	*note	13-01-LG	3.7.9	LCO		
3.7.12	APP		13-08-M	3.7.9	APP		
3.7.12	ACTION			3.7.9	ACTION	A	3.7-17
3.7.12	ACTION			3.7.9	ACTION	B.1	
3.7.12	ACTION		13-08-M	3.7.9	ACTION	B. 2	3.7-17
N/A				3.7.9.1	SR	Not Used	3.7-17
4.7.12	SR	a., b. & c.		3.7.9.2	SR		3.7-17
N/A				3.7.9.3	SR	Not Used	3.7-17
N/A				3.7.9.4	SR	Not Used	3.7-17
3.7.5.1	LCO		10-01-LG	3.7.10	LCO		
3.7.5.1	APP		10-02-M	3.7.10	APP		

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item.</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.7.5.1	ACTION	MODE 1-4		3.7.10	ACTION	A.1	
3.7.5.1	ACTION	MODE 1-4		3.7.10	ACTION	B.1	
3.7.5.1	ACTION	MODE 1-4		3.7.10	ACTION	B.2	
3.7.5.1	ACTION	MODE 5/6 a		3.7.10	ACTION	C.1	
3.7.5.1	ACTION	MODE 5/6 a	10-05-LS18	3.7.10	ACTION	C.2.1	
3.7.5.1	ACTION	MODE 5/6 a	10-05-LS18	3.7.10	ACTION	C.2.2	
3.7.5.1	ACTION	MODE 5/6b	10-06-LG, 10-16-LG	3.7.10	ACTION	D.1	
3.7.5.1	ACTION	MODE 5/6b	10-02-M	3.7.10	ACTION	D.2	
3.7.5.1	ACTION	New M. 1-4	10-04-A	3.7.10	ACTION	E	
4.7.5.1	SR	b.1)		3.7.10.1	SR		
4.7.5.1	SR	New	10-08-A	3.7.10.2	SR		
4.7.5.1	SR	e. 2)	10-10-TR1	3.7.10.3	SR		3.7-35
4.7.5.1	SR	e. 3)	10-11-LS19	3.7.10.4	SR		3.7-33
N/A					Not used.	3.7-52	
N/A						Not used.	
N/A						Not used.	
N/A						Not used.	
N/A						Not used.	
N/A						Not used.	

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item.</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
N/A						Not used.	
N/A						Not used.	
N/A						Not used.	
N/A						Not used.	
N/A						Not used.	
3.7.6.1	LCO			3.7.12	LCO		
3.7.6.1	APP			3.7.12	APP		
3.7.6.1	ACTION	a.		3.7.12	ACTION	A	3.7-21
3.7.6.1	ACTION	b.		3.7.12	ACTION	B	
N/A				3.7.12	ACTION	C Not Used	3.7-23
3.7.6.1	ACTION	a. & b.		3.7.12	ACTION	D.1	
3.7.6.1	ACTION	a. & b.		3.7.12	ACTION	D.2	
4.7.6.1	SR	a. 1)	12-06-LG	3.7.12.1	SR		
4.7.6.1	SR	New	10-08-A	3.7.12.2	SR		
4.7.6.1	SR	d. 2)	12-04-TR1	3.7.12.3	SR		3.7-22
N/A				3.7.12.4	Not Used		3.7-23
N/A				3.7.12.5	Not Used		3.7-24
4.7.6.1	SR	d. 4)		3.7.12.6	SR	New	3.7-18
3.9.12	LCO			3.7.13	LCO		
3.9.12	APP			3.7.13	APP		
				3.7.13	APP	NOTE	3.7-42
3.9.12	ACTION			3.7.13	ACTION	A.1	
3.9.12	ACTION	a.			ACTION	A.2	3.7-43

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.9.12	ACTION			3.7.13	ACTION	A.3	
N/A				3.7.13	ACTION	B Not Used	
N/A				3.7.13	ACTION	C Not Used	3.7-43
3.9.12		b.		3.7.13	ACTION	D	
4.9.12	SR	a.		3.7.13.1	SR		
4.9.12	SR	New		3.7.13.2	SR		
4.9.12	SR	d. 2)		3.7.13.3	SR		
4.9.12	SR	d. 3)		3.7.13.4	SR		3.7-49
N/A				3.7.14	Not Used		3.7-55
3.9.11	LCO			3.7.15	LCO		
3.9.11	APP			3.7.15	APP		
3.9.11	ACTION	b.		3.7.15	ACTION	NOTE	
3.9.11	ACTION	a.		3.7.15	ACTION	A	
4.9.11	SR			3.7.15.1	SR		
3.9.14.2	LCO			3.7.16	LCO		
3.9.14.2	APP			3.7.16	APP		3.7-53
3.9.14.2	ACTION	b.		3.7.16	ACTION	NOTE	
3.9.14.2	ACTION	a.		3.7.16	ACTION	A.1	
3.9.14.2	ACTION	a.		3.7.16	ACTION	A.2	3.7-53
Not Used				3.7.16	ACTION	A.2.2 Not Used	3.7-53
4.9.14.2	SR			3.7.16.1	SR		3.7-53

CROSS-REFERENCE TABLE FOR CTS 3/4.7
Sorted By Improved TS

<u>Current Specs</u>				<u>Improved Specs</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.9.14.3	LCO			3.7.17.1	LCO		3.7-51
3.9.14.3	LCO	a.		3.7.17.1	LCO	a.	3.7-51
3.9.14.3	LCO	b.		3.7.17.1	LCO	b.	3.7-51
3.9.14.3	APP			3.7.17.1	APP		3.7-51
3.9.14.3	ACTION	b.		3.7.17.1	ACTION	NOTE	3.7-51
3.9.14.3	ACTION	a.		3.7.17.1	ACTION	A	3.7-51
4.9.14.3	SR			3.7.17.1.1	SR		3.7-51
3.9-1	FIGURE			3.7.17-1	FIGURE		3.7-51
3.9.14.1	LCO			3.7.17.2	LCO		3.7-54
3.9.14.1	APP			3.7.17.2	APP		
3.9.14.1	ACTION	b.		3.7.17.2	ACTION	NOTE	
3.9.14.1	ACTION	a.		3.7.17.2	ACTION	A	3.7-54
4.9.14.1	SR			3.7.17.2.1	SR		3.7-54
3.9-2	FIGURE			3.7.17-2	FIGURE		
3.7.1.4	LCO			3.7.18	LCO		
3.7.1.4	APP			3.7.18	APP		
3.7.1.4	ACTION			3.7.18	ACTION	A.1	
3.7.1.4	ACTION			3.7.18	ACTION	A.2	
4.7.1.4	SR		04-01-M	3.7.18.1	SR		

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple

Methodology for Cross-Reference Tables
(Continued)

paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).

Methodology for Cross-Reference Tables
(Continued)

- Note: The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."
- Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up:

Specification

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Methodology

2 pages

3/4.7 PLANT SYSTEMS

3/4.7.1 ~~TURBINE CYCLE MAIN STEAM SAFETY VALVES (MSSVs)~~

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

01-01-A

APPLICABILITY: MODES 1, 2 and 3.

ACTION*:

- a. With one or more main steam line Code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced reduce power per Table 3.7-1; otherwise, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

01-02-LS1

01-04-LS3

(new) Reduce the Power Range High Neutron Flux reactor trip set points to less than or equal to the Maximum Allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs within 72 hours. Otherwise, be in at least MODE 3 within 6 hours and in MODE 4 within 12 hours.

01-04-LS3

- b. The provisions of Specification 3.0.4 are not applicable.

01-05-M

(new) With one or more steam generators with less than two MSSVs OPERABLE, be in at least MODE 3 within the next 6 hours and in MODE 4 within 12 hours.

01-06-M

SURVEILLANCE REQUIREMENTS

4.7.1.1** No additional requirements other than those required by Specification 4.0.6. Verify each required MSSV lift setpoint per Table 3.7-2 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ± 1%.

01-07-A

* Separate Condition entry is allowed for each MSSV.

01-02-LS1

** Only required to be performed in MODES 1 and 2.

01-05-M

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM
LINE SAFETY VALVES

01-04-LS3

MAXIMUM NUMBER OF INOPERABLE
SAFETY VALVES ON ANY OPERATING
STEAM GENERATOR

MAXIMUM ALLOWABLE POWER RANGE
NEUTRON FLUX HIGH SETPOINT
(PERCENT OF RATED THERMAL POWER)

1

87* ~~92~~

2

64* ~~45~~

01-09-LS31

3

42* ~~27~~

*Unless the Reactor Trip System breakers are in the open position.

01-04-LS3

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

<u>LIFT SETTING</u> [*]	<u>ORIFICE SIZE</u>	<u>01-10-LG</u>
1065 psig (-2%,+3%) ^{**}	4.515 inches	
1078 psig (+3%) ^{**}	4.515 inches	<u>01-11-EG</u>
1090 psig (+3%) ^{**}	4.515 inches	
1103 psig (+3%) ^{**}	4.515 inches	
1115 psig (+3%) ^{**}	4.515 inches	

~~* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.~~

01-10-EG

~~** Within +1% following main steam line Code Safety Valve testing.~~

01-07-A

PLANT SYSTEMS

AUXILIARY FEEDWATER (AFW) SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three ~~three~~ steam generator auxiliary feedwater pumps and ~~trains~~ associated flow paths shall be OPERABLE with:

02-01-LG

- a. ~~Two motor driven auxiliary feedwater pumps, each capable of being powered from separate vital busses, and~~
- b. ~~One steam turbine driven auxiliary feedwater pump capable of being powered from two OPERABLE and redundant steam supply sources.~~

APPLICABILITY: MODES 1, 2 and 3-

02-04-M

~~MODE 4* when steam generator is relied upon for heat removal~~

ACTIONS:

02-02-LS5

~~(new) With one steam supply to turbine driven AFW inoperable, restore the inoperable steam supply to OPERABLE status within 7 days and 10 days from discovery of failure to meet the LCO~~

02-03-M

a. ~~With one auxiliary feedwater pump inoperable in MODE 1, 2 or 3, restore the required auxiliary feedwater pumps train to OPERABLE status within 72 hours and within 10 days of discovery of failure to meet the LCO or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 12 hours.~~ or

02-03-M

02-10-LS21

b. ~~With two auxiliary feedwater pumps trains inoperable in MODE 1, 2 or 3, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 12 hours.~~

02-10-LS21

c. ~~With three auxiliary feedwater pumps trains inoperable in MODE 1, 2 or 3, immediately initiate corrective action to restore at least one auxiliary feedwater pump train to OPERABLE status as soon as possible. NOTE: LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status.~~

02-05-A

~~(new) With the required AFW train inoperable in MODE 4 immediately initiate action to restore the AFW train to OPERABLE status~~

02-04-M

~~*Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4~~

02-04-M

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump train shall be demonstrated OPERABLE:

a. At least once per 31 days by-

- 1) Deleted.

SURVEILLANCE REQUIREMENTS

- (2) Verifying that each ~~non-automatic manual power operated and automatic valve in the pump water flow path and both steam supplies to the steam turbine driven pump that is not locked, sealed, or otherwise secured in position, is in its correct position.~~ 02-07-M
02-09-A

- (3) ~~Verifying that each non automatic valve in both steam supplies to the steam turbine driven pump that is not locked, sealed, or otherwise secured in position, is in its correct position.~~ 02-09-A

- b. * ~~At least once per 92 days on a STAGGERED TEST BASIS by: Testing the steam turbine-driven pump and motor-driven pumps pursuant to in accordance with the frequency specified in Specification 4.0.5* the Inservice Testing Program. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the steam turbine-driven pump.~~ 02-08-LS6
02-04-M

- c. ** ~~At least once per 18 months by verifying that each auxiliary feedwater pump starts and valve opens* as designed automatically upon receipt of an a simulated or actual Auxiliary Feedwater Actuation test signal.~~ 02-11-A
02-12-TRI

- (new)
* ** ~~At least once per 18 months by verifying each AFW pump starts automatically on an actual or simulated actuation signal.~~ 02-11-A
02-12-TRI
02-04-M

- * ~~Not required to be performed for the steam turbine-driven pump, when until 24 hours after the secondary steam supply pressure is greater than 650 psig.~~ 02-14-M

- ** ~~Not applicable in MODE 4 when steam generator is relied upon for heat removal.~~ 02-04-M

PLANT SYSTEMS

AUXILIARY FEEDWATER SOURCES - CONDENSATE STORAGE TANK (CST) AND FIRE WATER STORAGE TANK (FWST)

LIMITING CONDITION FOR OPERATION

3.7.1.3 The Condensate Storage Tank (CST) ~~level~~ shall have a usable volume of ~~be at least 164,678 gallons of water 41.3%~~ with an open flow path to the Auxiliary 03-01-LG Feedwater (AFW) pump suction, and the Fire Water Storage Tank (FWST) ~~level~~ shall have a usable volume of ~~be at least 57,922 gallons of water 22.2%~~ for one Unit operation and ~~115,844 gallons of water 41.7%~~ for two Unit operation with a flow path capable of being aligned to the AFW pump suction.

APPLICABILITY: MODES 1, 2 and 3-~~4~~

~~MODE 4 when steam generator is relied upon for heat removal~~

02-04-M

ACTION:

- a. With the CST usable volume less than 164,678 gallons ~~level not within limits~~, or with the CST flow path not open to the AFW pumps suction, within four hours restore the required CST conditions; or, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN ~~without reliance upon steam generator for heat removal~~ within the following ~~6~~12 hours. 02-04-M
02-10-LS21
- b. With an FWST usable volume less than 57,922 gallons for one Unit operation and 115,844 gallons for two Unit operation ~~level not within limits~~, or with the FWST flow path not capable of being aligned to the AFW pump suction, within seven days restore the required FWST conditions; or, be in at least HOT STANDBY, within the next 6 hours and in HOT SHUTDOWN ~~without reliance upon steam generator for heat removal~~ within the following ~~6~~12 hours. 02-10-LS21
02-04-M

SURVEILLANCE REQUIREMENTS

- 4.7.1.3.1 The CST volume shall be demonstrated at least once per 12 hours by verifying the usable volume ~~level~~ is within its limits.
- 4.7.1.3.2 The FWST volume shall be demonstrated at least once per 12 hours by verifying the usable volume ~~level~~ is within its limits.
- 4.7.1.3.3 Verify the FWST is capable of being aligned to the Auxiliary Feedwater System by cycling each FWST valve in the flow path necessary for realignment through at least one full cycle once per quarter. 02-13-A

PLANT SYSTEMS

SECONDARY SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.1 microcurie/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.1 microcurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 ~~Verify that~~ The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1 ~~is~~ $\leq 0.10 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 once per 31 days.

04-01-M

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

<u>TYPE OF MEASUREMENT</u> <u>AND ANALYSIS</u>	<u>SAMPLE AND ANALYSIS</u> <u>FREQUENCY</u>	
1. Gross Radioactivity Determination hours.	At least once per 72	<u>04-02-LS8</u>
2. Isotopic Analysis for DOSE ¹³¹I Concentration	a) Once per 31 days, EQUIVALENT I whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines.	
	b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.	<u>04-01-M</u>

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES (MSIVs)

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve ~~Four~~ (MSIV) shall be OPERABLE.

05-04-A

APPLICABILITY: MODES 1, ~~MODES~~ 2 and 3 ~~except when all MSIVs are closed and de-activated.~~

05-01-LS9

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 48 hours; otherwise be in ~~HOT STANDBY~~ ~~MODE 2~~ within the next 6 hours and in ~~HOT SHUTDOWN~~ within the following 6 hours.

05-02-LS11

MODES 2 and 3*:

With one ~~or more~~ MSIVs inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed within 8 hours and verified closed once per 7 days. Otherwise, be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

05-02-LS11

05-05-M

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE** by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. ~~The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.~~

05-07-LS23

~~(new) Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal once every 18 months.~~

05-08-M

~~* Separate condition entry is allowed for each MSIV~~

05-02-LS11

~~** Only required to be performed in MODES 1 and 2~~

05-07-LS23

PLANT SYSTEMS

STEAM GENERATOR 10% ATMOSPHERIC DUMP VALVES (ADV'S)

LIMITING CONDITION FOR OPERATION

3.7.1.6 Four steam generator 10% atmospheric dump valves (ADV's) with the associated block valves open and associated remote manual controls, including the backup air bottles, ~~lines~~ shall be OPERABLE.

06-01-LG

06-09-A

APPLICABILITY: MODES 1, 2, and 3-

~~MODE 4 when steam generator is relied upon for heat removal~~

02-04-M

ACTION:

06-02-LS14

a. With one less than the required number of 10% ADV's ~~lines~~ OPERABLE, restore the inoperable steam generator 10% ADV ~~line~~ to OPERABLE status within 7 days ~~(304 is not applicable)~~; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN ~~without reliance upon steam generator for heat removal~~ within the following ~~612~~ hours.

02-10-LS21

02-04-M

b. With two less than the required number of 10% ADV's ~~lines~~ OPERABLE, restore at least one of the inoperable steam generator 10% ADV's to OPERABLE status within 72 hours; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN ~~without reliance upon steam generator for heat removal~~ within the following ~~612~~ hours.

02-10-LS21

02-04-M

~~(new) With three or more ADV lines inoperable, restore at least two to an OPERABLE status within 24 hours or be in MODE 3 in 6 hours and be in MODE 4 without reliance upon steam generators for heat removal within 18 hours~~

06-05-LS24

02-10-LS21

02-04-M

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each steam generator 10% ADV ~~lines~~, associated block valve and associated remote manual controls including the backup air bottles shall be demonstrated OPERABLE:

06-01-LG

a. At least once per 24 hours by verifying that the backup air bottle for each steam generator 10% ADV has a pressure greater than or equal to 260 psig, and

b. ~~At least once per 31 days by verifying that the steam generator 10% ADV block valves are open, and~~

06-06-LG

c. At least once per 18 months by verifying that all steam generator 10% ADVs will operate using the remote manual controls and the backup air bottles.

06-06-LG

~~(new) Verify one complete cycle of each 10% ADV block valve once per 18 months~~

06-04-M

PLANT SYSTEMS

MAIN FEEDWATER ISOLATION (MFIVS), REGULATING (MFRVS) AND ASSOCIATED BYPASS AND ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.7 ~~Four~~ In each feedwater line, each Main Feedwater Isolation Valve (MFIV)s shall be OPERABLE or closed. Each Main Feedwater Regulating Valve (MFRV)s, and ~~four~~ MFRV bypass valves shall be OPERABLE, closed or isolated.

07-01-A

APPLICABILITY: MODES 1, 2, and 3 except when MFIV, MFRV, or associated bypass valve is closed and de-activated or isolated by a closed manual valve.

07-02-LS37

ACTION:*

07-03-LS15

With one or more MFIVs, one or more MFRVs, or one or more MFRV bypass valves inoperable either:

07-06-LS33

- a. Restore the inoperable valve to OPERABLE status within 4 hours, or
- b. Close the inoperable valve within 4 hours, or
- c. If the inoperable valve is a MFRV or MFRV bypass valve, isolate the inoperable valve with at least one closed valve within 4 hours, or

07-04-LS16

07-04-LS16

07-04-LS16

(new) With two valves in the same flowpath inoperable, isolate the affected flowpath within 8 hours, or

07-04-LS16

- d. Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

(new) Verify the inoperable MFIV, MFRV, or MFRV bypass valve(s) closed or isolated once per 7 days.

07-11-M

SURVEILLANCE REQUIREMENTS

4.7.1.7.1 Each MFRV and MFRV bypass valve shall be demonstrated OPERABLE by determining the isolation time of each valve to be less than or equal to 7 seconds (not including instrument delays) at least each COLD SHUTDOWN but not more frequently than once per 92 days.

4.7.1.7.2 Each MFIV shall be demonstrated OPERABLE by determining the isolation time of each valve to be less than or equal to 60 seconds (not including instrument delays) when tested pursuant to Specification 4.0.5.

(new) Verify each feedwater isolation valve actuates to the isolation position on an actual or simulated actuation signal once every 18 months.

05-08-M

* Separate condition entry is allowed for each valve.

07-03-LS15

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PLANT SYSTEMS

3/4.7.3 VITAL COMPONENT COOLING WATER (CCW) SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two vital component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:*

08-02-A

With only one vital component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two vital component cooling water loops shall be demonstrated OPERABLE:

- a. ** At least once per 31 days by verifying that each CCW valve (manual, power-operated, or automatic) in the flow path servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position; and 08-04-A
08-08-A
- b. At least once per 18 months, by verifying that each automatic valve in the flow path that is not locked, sealed or otherwise secured in position servicing safety-related equipment actuates to its correct position on a Safety Injection or Phase "B" Isolation test an actual or simulated actuation signal, as appropriate. 08-08-A
08-05-A

(new) Verify each CCW pump starts automatically on an actual or simulated actuation signal at least once per 18 months. 08-06-TRI
08-07-M

* Enter applicable conditions and required actions of LCO 3, 4, 6, "RCS Loops" MODE 4, for residual heat removal loops made inoperable by CCW. 08-02-A

** Isolation of CCW flow to individual components does not render the CCW system inoperable. 08-04-A

PLANT SYSTEMS

3/4.7.4 AUXILIARY SALTWATER SYSTEM (ASW)

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two auxiliary saltwater trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION: ~~XXXX~~

08-02-A

With only one auxiliary saltwater train OPERABLE, restore at least two trains to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two auxiliary saltwater trains shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, ~~or power-operated, or automatic~~) in the flow path servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position or that power and/or air supplies are available such that the valve would be capable of being placed in its correct position.

09-03-A

09-01-M

(new) Verify each ASW pump starts automatically on an actual or simulated actuation signal at least once per 18 months.

09-04-M

(new) Verify each ASW power operated valve in the flow path that is not locked, sealed or otherwise secured in position is capable of being placed in the correct position in accordance with the Inservice Testing Program.

09-06-M

Enter applicable conditions and required actions of LCD 3.4.6, RCS Loops, MODE 4, for residual heat removal loops made inoperable by ASW/CCW.

08-02-A

PLANT SYSTEMS

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM (CRVS)

LIMITING CONDITION FOR OPERATION

~~3.7.5.1 The two Control Room Ventilation System* trains shall be OPERABLE** with two separate trains with each train consisting of one main supply fan, one filter booster fan, one pressurization supply fan and one HEPA Filter and Charcoal Adsorber System.~~

10-01-LG

10-13-LG

APPLICABILITY: All MODES.

~~During movement of irradiated fuel assemblies~~

10-02-M

ACTION:

MODES 1, 2, 3, and 4:

With one Control Room Ventilation System train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~With two trains of CRVS trains inoperable in MODES 1, 2, 3 or 4 immediately enter 3.0.3.~~

10-04-A

MODES 5 and 6 ~~or during movement of irradiated fuel assemblies:~~

10-02-M

a. With one Control Room Ventilation System train inoperable, restore the inoperable train to OPERABLE status within 7 days or immediately initiate and maintain operation of the OPERABLE Control Room Ventilation System train in the recirculation mode or immediately suspend CORE ALTERATIONS and suspend movement of irradiated fuel assemblies.

10-05-LS18

b. With both Control Room Ventilation System trains inoperable, or with the OPERABLE Control Room Ventilation System required to be in the recirculation mode by ACTION a. not capable of being powered by an OPERABLE emergency power source, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes or movement of irradiated fuel assemblies.

10-06-LG

10-16-LG

10-21-LS38

SURVEILLANCE REQUIREMENTS

4.7.5.1 Each Control Room Ventilation System train shall be demonstrated OPERABLE:-

a- ~~At least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F;~~

10-07-R

~~*The Control Room Ventilation System is common to both units.~~

10-02-M

~~**The system may be considered OPERABLE with no chlorine monitors, provided no bulk chlorine gas is stored within the SITE BOUNDARY.~~

10-13-LG

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (continued)

b. At least once per 31 days by:

- 1) ~~Initiating flow through the HEPA Filter And Charcoal Adsorber System and verifying that either redundant set of booster and pressurization supply fans the system operates for at least 10 continuous hours with the heaters operating.~~ 10-06-LG
- 2) ~~Verifying that each Ventilation System redundant fan is aligned to receive electrical power from a separate OPERABLE vital bus, and~~ 10-16-LG
- 3) ~~Starting (unless already operating) each main supply fan, booster fan, and pressurization supply fan, and verifying that it operates for 1 hour.~~ 10-06-LG

~~(new) Perform required CRVS FILTER TESTING in accordance with the Ventilation Filter Testing Program (VFIP).~~ 10-08-A

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

- 1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in ANSI N510-1980, and the system flow rate is 2100 cfm \pm 10%; 10-08-A
- 2) ~~Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D 3803-1989 at a temperature of 30°C and at 70% R.H. for a methyl iodide penetration of less than 1%; and~~ 10-23-LS13
10-24-A
- 3) ~~Verifying a system flow rate of 2100 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1980.~~ 10-17-A

d. After 720 hours of charcoal adsorber operation, by verifying, ~~within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D 3803-1989 at a temperature of 30°C and at 70% R.H. for a methyl iodide penetration of less than 1%;~~ 10-08-A
10-23-LS13
10-24-A

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- e. At least once per 18 months by: 10-08-A
- 1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3.5 inches Water Gauge while operating the system at a flow rate of 2100 cfm \pm 10%;
 - ~~2) Verifying that on an actual or simulated actuation a Phase "A" Isolation test signal, the system automatically switches into the pressurization mode of operation with approximately 27% (determined by damper position) of the flow through the HEPA filters and charcoal adsorber banks;~~ 10-10-IR1
10-08-A
 - ~~3) Verifying on a STAGGERED TEST BASIS that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during the pressurization mode of system operation; and~~ 10-11-LS19
 - 4) Verifying that the heaters dissipate 5 ± 1 kW when tested in accordance with ANSI N510-1980. 10-08-A
- f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 or a DOP test aerosol while operating the system at a flow rate of 2100 cfm \pm 10%; and 10-08-A
- g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon test gas while operating the system at a flow rate of 2100 cfm \pm 10%. 10-08-A

PLANT SYSTEMS

3/4.7.6 AUXILIARY BUILDING SAFEGUARDS AIR FILTRATION VENTILATION SYSTEM (ABVS)

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two Auxiliary Building Safeguards Air Filtration Ventilation System exhaust trains with one common HEPA filter and charcoal adsorber bank and at least two supply and exhaust fans shall be OPERABLE.

12-10-M

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

- a. With the HEPA filter and charcoal adsorber bank inoperable, restore the HEPA filter and charcoal adsorber bank to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With only one supply and one exhaust fan OPERABLE, restore at least two supply and two exhaust fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

12-10-M

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each Auxiliary Building Safeguards Air Filtration Ventilation System train shall be demonstrated OPERABLE:

- a. At least once per 31 days by:
 - 1) ~~Initiating flow through the HEPA filter and charcoal adsorber bank and verifying that the train operates for at least 10 continuous hours with the heaters operating, and~~
 - 2) ~~Verifying that each exhaust fan is aligned to receive electrical power from a separate OPERABLE vital bus.~~

12-06-LG

12-06-LG

~~(new) Perform required ABVS filter system testing in accordance with the Ventilation Filter Testing Program (VFETP).~~

10-08-A

- b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by:

10-08-A

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- 1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in ANSI N510-1980, and the system flow rate is 73,500 cfm \pm 10%;
- 2) Verifying, ~~within 31 days after removal,~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D 3803-1989 at a ~~temperature of 30 °C and at~~ 70% R.H. for a methyl iodide penetration of less than 6%; and 10-23-LS13
10-24-A
- 3) Verifying a system flow rate of 73,500 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1980. 10-17-A
10-08-A
- c. After every 720 hours of charcoal adsorber operation, by verifying, ~~within 31 days after removal,~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-1989 at a ~~temperature of 30 °C and at~~ 70% R.H. for a methyl iodide penetration of less than 6%. 10-23-LS13
- d. At least once per 18 months by:
 - 1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3.7 inches Water Gauge while operating the system at a flow rate of 73,500 cfm \pm 10%. 10-08-A
 - 2) Verifying that ~~each train actuates and that flow is established through the HEPA filter and charcoal adsorber bank on a Safety Injection test~~ ~~an actual or simulated actuation~~ signal, and 12-10-M
12-04-IR1
 - 3) Verifying that the heaters dissipate 50 \pm 5 kW when tested in accordance with ANSI N510-1980. 10-08-A
 - 4) Verifying that leakage through the Auxiliary Building Safeguards Air Filtration System Dampers M2A and M2B is less than or equal to 5 cfm when subjected to a Constant Pressure or Pressure Decay Leak Rate Test in accordance with ASME N510-1989. The test pressure for the leak rate test shall be based on a maximum operating pressure as defined in ASME N510-1989, of 8 inches water gauge.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 73,500 cfm \pm 10%; and 10-08-A
- f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon test gas while operating the system at a flow rate of 73,500 cfm \pm 10%. 10-08-A

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PLANT SYSTEMS

PLANT SYSTEMS

3/4.7.12 ULTIMATE HEAT SINK (UHS)

LIMITING CONDITION FOR OPERATION

3.7.12 The ultimate heat sink (UHS)* shall be OPERABLE with an inlet water temperature of less than or equal to 64°F.

13-01-LG

APPLICABILITY: MODES 1, 2, and 3 and 4.

13-08-M

ACTION:

With the requirements of the above specification not satisfied, place a second vital component cooling water heat exchanger in service within 8 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT/COLD SHUTDOWN within the following 636 hours. ~~The provisions of Specification 3.0.4 are not applicable.~~

13-08-M

13-09-A

SURVEILLANCE REQUIREMENTS

4.7.12 The UHS shall be determined OPERABLE by verifying the inlet water temperature to be within its limit:

- a. At least once per 24 hours when the inlet water temperature is equal to or less than 60°F, or
- b. At least once per 12 hours when the inlet water temperature is greater than 60°F but less than 62°F, or
- c. At least once per 2 hours when the inlet water temperature is equal to or greater than 62°F but less than or equal to 64°F.

~~*The UHS is common to both units.~~

13-01-LG

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. Deletions - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. Additions - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. Modifications - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. Administrative - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

**Methodology For Mark-Up of Current TS
(Continued)**

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers	2 Pages
Description of Changes	13 Pages

TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS

SECTION 3/4.7

NUREG-1431 Title	CHG NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Main Steam Safety Valves	01	3.7.1.1	3.7.1.1	3.7.1.1	3.7.1.1
Auxiliary Feedwater System	02	3.7.1.2	3.7.1.2	3.7.1.2	3.7.1.2
Condensate Storage Tank	03	3.7.1.3	3.7.1.3	3.7.1.3	3.7.1.3
Secondary Specific Activity	04	3.7.1.4	3.7.1.4	3.7.1.4	3.7.1.4
Main Steam Isolation Valves	05	3.7.1.5	3.7.1.5	3.7.1.5	3.7.1.5
Atmospheric Dump Valves	06	3.7.1.7	3.7.1.6	3.7.1.7	3.7.1.6
Main Feedwater Isolation Valves & Main Feedwater Regulation Valves [and Associated Bypass Valves]	07	3.7.1.6	3.7.1.7	3.7.1.6	3.7.1.7
Component Cooling Water System	08	3.7.3	3.7.3	3.7.3	3.7.3
Service Water System	09	3.7.4	3.7.4	3.7.4	3.7.8
Control Room Emergency Filtration System	10	3.7.6	3.7.6	3.7.7.1	3.7.5
Control Room Emergency Air Temperature Control System	11	N/A	N/A	3.7.7.2	N/A
Emergency Core Cooling System Pump Room Exhaust Air Cleanup System	12	N/A	N/A	3.7.8	3.7.6
Fuel Building Air Cleanup System	12	3.7.7	3.7.7	N/A	N/A
Ultimate Heat Sink	13	3.7.5	3.7.5	3.7.5	3.7.12
Steam Generator Pressure/Temperature Limitation	14	N/A	N/A	3.7.2	N/A
Flood Protection	15	N/A	N/A	3.7.6	N/A
Snubbers	16	N/A	N/A	3.7.9	N/A
Area Temperature Monitoring	17	N/A	N/A	3.7.10	N/A
Main Feedwater Pressure/Temperature Limitation	18	N/A	N/A	3.7.13	N/A

NUREG-1431 Title	CHG NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Safety Chilled Water System	19	N/A	N/A	3.7.12	N/A
UPS HVAC system	20	N/A	N/A	3.7.11	N/A

DESCRIPTION OF CHANGES TO TS SECTION 3/4.7

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	A	A reference to Table 3.7-2 is deleted from the limiting condition of operation (LCO) and moved to the surveillance requirement (SR) (refer to change 01-07-A). This change is consistent with NUREG-1431.
01-02	LS1	A Note is added to allow separate entry for each main steam safety valve (MSSV). The current specification requires that all MSSVs discovered to be inoperable be returned to OPERABLE within the initial four hours provided for the first inoperable MSSV. For example, if a second inoperable MSSV were discovered three hours into the ACTION statement of the first inoperable MSSV, both MSSVs would have to be returned to OPERABLE within the following one hour or the plant power level would have to be reduced to that specified in Table 3.7-1. The addition of the Note allowing separate entry for each inoperable MSSV provides the full four hours for each inoperable MSSV.
01-03	LS2	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-04	LS3	The CTS allows continued operation with inoperable MSSVs if the power range neutron flux high trips are reduced. [] Industry traveler WOG-83, Rev. 0 and draft Rev. 1, provided revised ACTIONS to require that: 1) the reactor power be reduced to compensate for the loss of pressure relief capacity to a maximum allowable power determined in accordance with Westinghouse NSAL 94-001 and NRC Information Notice 94-60, 2) the power range neutron flux high trip setpoint be reduced for inoperable MSSVs if a positive moderator temperature coefficient (MTC) exists at the allowed percent rated thermal power in MODE 1, and 3) the power range neutron flux high trip setpoints be reduced to account for a control rod withdrawal at partial power with more than one MSSV inoperable. In addition, the Completion Time for resetting the high flux trips is revised from four hours to 72 hours and the ACTION is revised to specifically require an appropriate power reduction within four hours. This is a relaxation since the CTS require the high neutron flux trip setpoint to be reduced as required within four hours for inoperable MSSVs regardless of the MTC value. Pending approval of draft Rev. 1 of WOG-83, the changes proposed in the traveler have been modified to retain

CHANGE
NUMBER

NSHC

DESCRIPTION

the CTS requirement to reset the power range neutron flux-high trip setpoints based on the number of MSSVs inoperable to a maximum allowable power determined in accordance with calculations or analysis to account for Westinghouse NSAL 94-001 and NRC Information Notice 94-60. However, the Completion Time of 72 hours proposed by WOG-83 has been retained and is justified based on the low probability of an event occurring during this time and the need to provide sufficient time to reset the channels in an orderly manner without inducing a transient due to human error. Retention of the CTS requirement for resetting the reactor trip setpoints is acceptable because this requirement is more conservative than the ACTIONS specified by either the ISTS or WOG-83, as revised. []

01-05

M

The exception to TS 3.0.4 is no longer needed due to the note associated with the revised surveillance. The exception was allowed to TS 3.0.4 due to the fact that the applicable MODES must be entered in order to perform the required surveillance (if the MSSVs are tested in place) and to allow Mode changes to be made if the applicable action was met. In the CTS, MODE 1, 2, or 3 could be entered. In NUREG-1431, the surveillance is modified by a more restrictive note that specifies that the surveillance need only be current prior to reaching MODE 2. The surveillance note still allows MODE changes into the MODE of Applicability of the LCO, i.e., MODE 3 for testing purposes.

01-06

M

The new ACTION adds an explicit requirement to be in MODE 3 in 6 hours and MODE 4 in 12 hours if any steam generator (SG) loop has less than two MSSVs OPERABLE. NUREG-1431 requires that the plant only be placed in a MODE where the specification is no longer applicable, which in this case would be MODE 4. The CTS would require the plant to enter TS 3.0.3 because operation with less than two MSSVs OPERABLE per SG is an undefined condition, and thus not permitted. Therefore, the new ACTION eliminates the one hour allowed for action via TS 3.0.3. This requirement is more restrictive with the loss of the one hour for ACTIONS required by TS 3.0.3.

01-07

A

The CTS SR is revised to specifically reference the In-service Testing (IST) Program developed per TS 4.0.5 and contained in the Administrative Section of the ISTS. The surveillance directly references Table 3.7-2 for lift points and incorporates the requirement that the MSSV as-left liftpoints to be within +/- 1 percent of the nominal setpoint.

01-08

Not Used.

**CHANGE
NUMBER**

NSHC

DESCRIPTION

01-09	LS31	The maximum power range neutron high flux trip setpoints required for one or more inoperable MSSVs are revised in accordance with the recommendations of Westinghouse NSAL 94-001, dated January 20, 1994, and a specific calculation and analysis to support the proposed revision. Since the issuance of NSAL 94-001, administrative controls have been in place to require a reactor power and high neutron flux trip setpoint reduction to be consistent with those values determined by NSAL 94-001 or as required by the CTS if they were more conservative. Since the trip setting of the CTS Table 3.7-1 is revised upward for one inoperable MSSV for incorporation into the ITS, this change has been classified as less restrictive. The settings for two or three inoperable MSSVs are more restrictive than the CTS or NUREG-1431 as modified by WOG-83.
01-10	LG	The note on Table 3.7-2 stating that the set pressures of the MSSVs shall correspond to the ambient conditions of the valve at normal operating temperatures is moved to the Bases of ITS SR 3.7.1.1. This change is acceptable because it removes details from the TS that are not required to protect the health and safety of the public while retaining the basic limiting condition for operation.
01-11	LG	The MSSV line orifice size is moved from Table 3.7-2 to a licensee-controlled document. This is design information that is not required in the ITS for operating or OPERABILITY concerns.
01-12	A	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
02-01	LG	The descriptive material related to the definition of an auxiliary feedwater (AFW) train is deleted from the LCO and moved to the Bases. This change is acceptable because it removes details from the TS that are not required to protect the health and safety of the public while retaining the basic limiting condition for operation.
02-02	LS5	The ACTION specifies the requirements for allowed outage time (AOT) should one of the steam supplies to the turbine driven auxiliary feedwater (TDAFW) pump become inoperable. A previous interpretation required that the TDAFW pump be declared inoperable and the ACTION statement for one inoperable pump be entered. This revision is a relaxation of the CTS requirements.
02-03	M	The ACTIONS are modified to require restoration of the systems to meet the LCO within 10 days of discovery of failure to meet the LCO. This new requirement is intended to prevent multiple overlapping ACTION entries such that the intended AOT is exceeded. This proposed change is acceptable because it results in more stringent TS requirements that are both appropriate and consistent with NUREG-1431.

**CHANGE
NUMBER**

NSHC

DESCRIPTION

02-04	M	The LCO Applicability and ACTION are revised to include a requirement that an AFW train with a motor driven pump, the appropriate atmospheric dump valves (ADV) and the condensate storage tank (CST) be operable in MODE 4 when the SG is relied upon for heat removal. This reflects the situation where credit is taken for a reactor coolant system (RCS) loop for heat removal in MODE 4 in lieu of a residual heat removal (RHR) train. (Refer to RCS HOT SHUTDOWN CTS.) This change clarifies the Applicability of AFW, CST and ADVs to the RCS loops - MODE 4 TS.
02-05	A	Although previously implied, the addition of the footnote for three inoperable AFW trains assures that TS 3.0.3 will not be entered and that no other ACTION statement for other inoperable components will be applied that might force the unit into an unsafe condition.
02-06	LG	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-07	M	The surveillance to verify valve alignments is revised to include automatic valves. This proposed change is acceptable because it results in more stringent TS requirements that are both appropriate and consistent with NUREG-1431.
02-08	LS6	The surveillance interval for the AFW pump performance is changed from once per 92 days on a STAGGERED TEST BASIS (STB) to "in accordance with the IST Program." This proposed change will eliminate any potential ambiguity associated with AFW pump testing as a result of ASME changes and results in consistent presentation of pump testing throughout the TS.
02-09	A	The surveillance is revised to combine the water flow and steam flowpaths into one SR. This proposed change is acceptable because it is administrative in nature; no change to technical requirements would result.
02-10	LS21	The time to achieve HOT SHUTDOWN conditions if ACTIONS are not completed is changed from 6 to 12 hours. This time is reasonable to reach the required conditions under the circumstances, since the SG heat removal system is degraded, but is the system performing the cooldown to MODE 4.
02-11	A	The testing requirements for pumps and valves are separated into two SRs in accordance with NUREG-1431.
02-12	TR1	The respective valve and pump SRs are revised to allow credit for an actual actuation, if one occurs, to satisfy the SRs. The identification of the initiating signal is moved to the Bases.
02-13	A	The requirement to verify that the fire water storage tank is capable of realignment as an AFW backup supply is moved to the AFW ITS from the CTS describing the AFW sources and volume requirements. Both the flow path verification and the frequency are moved to make the AFW ITS complete.

**CHANGE
NUMBER**

NSHC

DESCRIPTION

02-14	M	The Note for testing of the steam TDAFW pump is revised to explicitly define when the testing must be performed. Previously, the time period was unspecified.
02-15	LG	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
02-16	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
02-17	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
02-18	M	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
02-19	LG	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
02-20	LS35	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
02-21	LG	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
03-01	LG	The description of the AFW supply pathway is deleted due to its relocation to AFW ITS 3.7.5 and associated Bases. The revised specification deals only with the required volumes of the AFW pump supply sources.
03-02	LS22	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
03-03	LG	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
04-01	M	The revision requires isotopic analysis for DOSE EQUIVALENT I-131 concentration to be performed on a 31-day frequency. The conditional performance requirements in the current surveillance (i.e., based on 10 percent of the allowable limit for radionuclides contained in Table 4.7-1) are deleted. This results in a more restrictive requirement for the isotopic analysis.
04-02	LS8	The SR of Table 4.7-1 to determine gross radioactivity is deleted. The change is acceptable because radioiodines and the resulting thyroid dose are limiting, not noble gases and whole-body dose. The primary to secondary leakage limits and DOSE EQUIVALENT I-131 limits ensure the dose analyses in the [Final Safety Analysis Report] remain valid.
05-01	LS9	The MODE of Applicability for the main steam isolation valves (MSIVs) is revised to clarify that in MODES 2 and 3 with all MSIVs closed [and deactivated], the safety function of the MSIVs is already met. Thus, OPERABILITY of the MSIVs in this condition is not required.

**CHANGE
NUMBER**

NSHC

DESCRIPTION

05-02	LS11	This change revises the AOT/Completion Time for an inoperable MSIV from four hours to eight hours in MODE 1, 2, and 3. This is a relaxation from the AOT of the CTS which requires that the valve be returned to an OPERABLE condition within four hours and implies that the valve must be immediately closed in MODES 2 and 3. The revised completion time is acceptable due to the low probability of an accident occurring that would require the closing of the MSIVs. In addition, from a containment isolation standpoint, the completion time can be greater for these valves as opposed to other containment isolation valves because these valves isolate a closed system that penetrates containment and do not act directly as the containment boundary. This revision deletes the MODE 1 ACTION to place the plant in MODE 3, and then MODE 4 due to an inoperable MSIV and only requires entry into MODE 2 since the ACTIONS for MODE 2 or 3 operation would then be applicable. Operation in MODES 2 and 3 is revised to allow more than one MSIV to be inoperable and by a note to allow a separate condition entry for each inoperable MSIV.
05-03	LS12	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
05-04	A	The LCO is changed from the OPERABILITY of each MSIV to the OPERABILITY of four MSIVs.
05-05	M	A new requirement for a 7 day periodic verification of the closure [or isolation] of inoperable isolation valves is added to be consistent with NUREG-1431.
05-06		Not used.
05-07	LS23	A footnote is added to the SR to indicate that demonstration of MSIV OPERABILITY is only required to be performed for entry into (and continued operation in) MODES 1 and 2. The footnote is added in lieu of the current exception to CTS 4.0.4. While the footnote is intended to establish the same exemption, it is in fact less restrictive because the footnote permits an indefinite stay in MODE 3 while the exception to CTS 4.0.4 requires testing within 24 hours of establishing the necessary plant conditions per TS 4.0.3 as described in Generic Letter (GL) 87-09.
05-08	M	This change creates a new SR for the MSIVs and [main feedwater isolation valves (MFIVs)] to distinguish between the IST and the automatic actuation testing of these isolation valves. The surveillance allows credit for an actual actuation, if one occurs, to satisfy the SRs. This change is consistent with WOG-98. This proposed change is acceptable because it results in more stringent TS requirements that are both appropriate and consistent with NUREG-1431.
06-01	LG	This change revises the CTS to delete the descriptive material from the CTS LCO and moves this material to the Bases.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
06-02	LS14	This change adds an exception to LCO 3.0.4 for the 7 day ACTION to restore the atmospheric [dump] valve OPERABILITY. This change allows the plant to change MODES if one atmospheric [dump] valve is found inoperable while in MODE 2 or 3. Allowing MODE transition with an inoperable atmospheric [dump] valve does not significantly increase the risk since the remaining valves are OPERABLE.
06-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B)
06-04	M	A surveillance is added that requires the manual cycling of the atmospheric [dump] valve [block] valves [every 18 months]. This proposed change is acceptable because it results in more stringent TS requirements that are both appropriate and consistent with NUREG-1431.
06-05	LS24	This change adds a new ACTION for three or more inoperable atmospheric [dump] valves that requires action within 24 hours. The CTS would require entry into TS 3.0.3 for three inoperable atmospheric [dump] valves. However, NUREG-1431 recognizes the low probability of an accident requiring the atmospheric [dump] valves, thus permitting this configuration.
06-06	LG	This change moves the requirements for the surveillances to the Bases. The details of the specific test requirements such as those dealing with the "remote manual controls" are not included in the STS, but are included in the Bases section for the SR. [Verification that the block valves to the 10% ADVs are open is moved to the Bases as a procedural surveillance.] This change is acceptable because it removes details from the TS that are not required to protect the health and safety of the public while retaining the basic limiting condition for operation.
06-07	LS25	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
06-08	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
06-09	A	This change revises the LCO to refer to the atmospheric [dump] valve lines versus atmospheric [dump] valves.
06-10	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-01	A	The LCO is changed from the OPERABILITY of each main feedwater line to the OPERABILITY of four feedwater flow path isolation valves [and associated bypasses] [and the requirement to have the valves closed or isolated if not OPERABLE is moved to Applicability. The ACTIONS are also revised to compensate for the line requirement deletion].
07-02	LS37	The LCO Applicability is revised to exclude [MFIVs or FRVs] that are closed [and de-activated or isolated by a closed manual valve]. NUREG-1431 recognizes that if one or more [MFIVs or FRVs] or associated bypass valves are closed [and de-activated] and verified closed, their safety function is being fulfilled and there is no need to enter the ACTION statement.

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DESCRIPTION

07-03	LS15	The CTS are written on a per feedwater line basis and NUREG-1431 is written for four lines. A Note is added to the ACTION statement that allows separate condition entry for each inoperable valve. The addition of the Note allowing separate entry for each valve inoperable in a flow path provides the full AOT for each of the inoperable flow paths.
07-04	LS16	The AOT is revised for an individual inoperable valve from 4 hours to 72 hours. With one valve inoperable in the flow path, flow path isolation is still available. The extended AOT is based upon the ability to isolate the flow path and the low probability of the occurrence of a main steam line break accident. The ACTIONS are further revised to add a new ACTION for two inoperable valves in the same flowpath to address the loss of isolation capability. This new ACTION requires that the inoperable flow path be isolated within eight hours. The CTS require entry into TS 3.0.3 if two valves in a flow path are inoperable since the condition is undefined.
07-05	LS17	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-06	LS33	This change revises the ACTION to apply to one "or more" [MFIVs] inoperable, consistent with NUREG-1431. This is less restrictive than the CTS which applied to only one inoperable valve. The proposed change is acceptable due to the low probability of an event for which the [MFIVs] are required to be OPERABLE and the availability of alternate methods of performing the required function.
07-07	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-08	LS26	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-09	LS10	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-10	M	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-11	M	A new requirement for a 7 day periodic verification of the closure [or isolation] of inoperable isolation valves is added. This proposed change is acceptable because it results in more stringent TS requirements that are both appropriate and consistent with NUREG-1431.
07-12	LS4	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-13		Not used.
07-14		Not used.
07-15		Not used.
07-16	LS34	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
07-17	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
08-01	LG	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

DCPP Description of Changes to Current TS

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NUMBER**

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DESCRIPTION

08-02	A	A Note is added to the ACTION that references a potential interaction with ITS 3.4.6 dealing with OPERABILITY of the RHR system in MODE 4. The Note requires that the applicable TS be entered for the RHR train made inoperable by the inoperable [component cooling water (CCW) or auxiliary saltwater (ASW)] system. The ACTIONS of the referenced TS (RCS Loops-MODE 4) require more immediate action than are required by the [CCW or ASW] ACTIONS.
08-03	LG	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	A Note is added to the [CCW] surveillance that clarifies that the system is not made inoperable by the isolation of individual components. This change is in accordance with NUREG-1431, and provides clarification only.
08-05	A	The surveillance is modified to clarify that valves that are locked, sealed, or otherwise secured in their correct position are not required to be tested. This change is in accordance with NUREG-1431, and provides clarification only.
08-06	TR1	The SR is revised to allow credit for an actual actuation, if one occurs, to satisfy the SRs. The identification of the signal is moved to the Bases.
08-07	M	A new surveillance is added that requires verifying that each CCW pump starts automatically on an actual or simulated signal actuation at least once per 18 months.
08-08	A	The surveillances are revised to clarify that only verification of the correct position of valves in the flow path is required.
08-09		Not used.
08-10	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
09-01	M	This change revises the existing surveillance to verify that a motive source is available that would permit the required ASW valves to be repositioned. This change is consistent with the intent of NUREG-1431.
09-02	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
09-03	A	The surveillance is reworded and the requirement to verify the position of automatic valves is deleted since the ASW system has no automatic valves.
09-04	M	A new surveillance is added that requires verifying that each ASW pump starts automatically on an actual or simulated actuation signal at least once per 18 months.
09-05	TR1	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
09-06	M	A new surveillance is added that requires verifying that each valve in the flow path that is not locked, sealed, or otherwise secured in position is capable of being placed in the correct position in accordance with the Inservice Testing (IST) Program.

DCPP Description of Changes to Current TS

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
09-07	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-01	LG	The descriptive material, related to the definition of a ventilation train, is deleted from the LCO and moved to the Bases.
10-02	M	The Applicability and applicable ACTIONS are revised to incorporate "during movement of irradiated fuel assemblies" in addition to all MODES (i.e., MODES 1-6).
10-03	LS7	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-04	A	A new ACTION statement is added by NUREG-1431 to require entering TS 3.0.3 immediately if two trains of the CR ventilation system are inoperable in MODES 1, 2, 3, or 4. The CTS requires entry into TS 3.0.3, since the condition of two trains inoperable is undefined; therefore, the revision has been classified as administrative.
10-05	LS18	A new option is added to the ACTION by NUREG-1431 that allows the suspension of CORE ALTERATIONS or movement of irradiated fuel versus placing the CR ventilation system in the recirculation mode.
10-06	LG	The details and description of the required ACTIONS and the monthly SRs for train operability are moved to the Bases. This is an example of removing details that are not required to be in the TS and is consistent with NUREG-1431.
10-07	LG	The surveillance that verifies CR temperature once per 12 hours is moved to a licensee controlled document.
10-08	A	The description of the ventilation filter specific testing requirements and the required surveillances are moved to the Ventilation Filter Testing Program (VFTP) as defined in the Administrative Controls of the ITS. No technical changes to requirements or test specifics except as noted in separate change numbers are made. A new SR is added that requires [CR and Auxiliary Building ventilation] system filter testing in accordance with the VFTP. The requirements of this specification are: 1) moved to Section 5.5.11 of the ITS, or 2) deleted since they are duplicated in Regulatory Guide (RG) 1.52, Revision 2, [ANSI N510-1980, or ASTM D 3803-1989].
10-09	LS27	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-10	TR1	The SR is revised to allow credit for an actual actuation, if one occurs, to satisfy the SRs. The identification of the initiating signal is moved to the Bases.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
10-11	LS19	The Frequency of the surveillance requiring verification of the CR ventilation system capability to maintain a positive pressure is relaxed to 18 months on a STB, consistent with NUREG-1431. The new Frequency requires one of the two trains to be tested every 18 months instead of both trains every 18 months. The most likely cause of a failure to achieve the required pressure is a failure of the ventilation pressure boundary. Thus, when one train successfully demonstrates the ability to maintain the pressure, in all likelihood the other train will also. This results in less testing of the CR ventilation system than is required by the CTS.
10-12	LS32	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
10-13	LG	The footnotes indicating that CR ventilation system is common to both units, and that the system may be considered OPERABLE with no chlorine monitors if no bulk chlorine gas is stored within the SITE BOUNDARY, are moved to the Bases.
10-14	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
10-15	LG	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
10-16	LG	The requirement to have an available emergency power source for the CR ventilation system is moved to the Bases. This change is consistent with the Bases of NUREG-1431 for ventilation systems required during fuel movement.
10-17	A	The SR to measure ventilation system flow rate is not identified as a separate SR in the ITS because it is verified during the other in-place filter tests (see ITS 5.5.11).
10-18	LS36	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
10-19	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
10-20	LS39	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
10-21	LS38	The ACTION to immediately suspend all operations involving CORE ALTERATIONS and movement of irradiated fuel assemblies when both trains of CR ventilation are inoperable in MODES 5 and 6 and during movement of irradiated fuel assemblies is deleted consistent with NUREG-1431. This change is acceptable because the immediate suspension of CORE ALTERATIONS and movement of irradiated fuel provides adequate protection from a release of radioactivity. Boron dilution events leading to criticality are not postulated as these events are prevented from occurring.
10-22	M	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).

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DESCRIPTION

10-23	LS13	The "within 31 days after removal" requirement for completion of laboratory analyses is deleted. This requirement is not contained in the ITS nor is it contained in the RG 1.52 or the applicable ANSI standards. Failure to complete an analysis within 31 days has insignificant safety consequences because the results would be available within approximately the same time period and it is very unlikely that the charcoal would be degraded to the extent that there would be a complete loss of a safety function.
10-24	A	The 30°C temperature specified for laboratory testing of filter carbon samples is added to be consistent with NUREG-1431. [This temperature is also specified in ASTM D 3803-1989.] This proposed change is acceptable because it adds a requirement already required to be performed when testing in accordance with the standard already specified in the TS; no change to technical requirements would result.
11-01	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
11-02	LS28	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-01	LS29	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-02	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-03		Not used.
12-04	TR1	The SR is revised to allow credit for an actual actuation, if one occurs, to satisfy the SRs. The identification of the specific signals that initiate the change in state of the equipment is moved to the Bases.
12-05	LS32	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-06	LG	The descriptive material and details of the retained ventilation surveillance testing are moved to the Bases. The change is acceptable because it removes detail no longer required in the TS to protect the health and safety of the public while retaining the basis limiting condition for operation.
12-07	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-08	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-09	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-10	M	The CTS ABVS LCO is extended to the supply fans. The ABVS "train" is defined in the ITS Bases to include the supply fans.
13-01	LG	The Note stating that the ultimate heat sink (UHS) is common to both units is moved to the Bases, consistent with NUREG-1431.
13-02	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
13-03	R	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

DCPP Description of Changes to Current TS

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
13-04	LG	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B)
13-05	LS30	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
13-06	LG	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
13-07	LG	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
13-08	M	The Applicability is extended to MODE 4 to be consistent with NUREG-1431 and the MODE 4 OPERABILITY requirements associated with CCW and ASW. The ACTION is revised to require entering COLD SHUTDOWN within 36 hours which is consistent with CTS 3.0.3 and NUREG-1431. This proposed change is acceptable because it results in more stringent TS requirements that are both appropriate and consistent with NUREG-1431.
13-09	A	The exemption to TS 3.0.4 is deleted as it is no longer needed. The revised LCO 3.0.4 of NUREG-1431 is not Applicable if entry into a Required Action allows indefinite operation in that ACTION. Placing the second CCW heat exchanger in service allows operation to continue indefinitely.
14-01	R	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
15-01	R	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
16-01	R	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
17-01	R	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
18-01	A	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
18-02	LG	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
18-03	TR1	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
19-01	R	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
20-01	LG	Not applicable to DCPD - see Conversion Comparison Table (Enclosure 3B).
20-02	TR1	Not applicable to DCPD - see Conversion Comparison Table (Enclosure 3B).

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(21 pages)

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	Reference to Table 3.7-2 is deleted from the LCO and moved to the SR (refer to 01-07-A).	Yes	Yes	Yes	Yes
01-02 LS1	A Note is added to allow separate condition entry for each MSSV which allows the full four hours for each inoperable MSSV.	Yes	Yes	Yes	Yes
01-03 LS2	This CPSES specific revision relaxes the as-found MSSV lift tolerances from +/- 1% to +/- 3%.	No, LA 108/107 issued 10/1/95 to relax setpoint (refer also to 01-13-LS20).	Yes	No	No
01-04 LS3	Revised ACTIONS for inoperable MSSVs: 1) specifically requires a power reduction within four hours and 2) requires the reactor power neutron flux high trip set point to be reduced within 72 hours.	Yes	Yes	Yes	Yes
01-05 M	The ACTION of the CTS which allowed an exception to TS 3.0.4 is deleted due to the Note associated with revised SR 4.7.1.1, which allows a MODE change into MODE 3, one of the MODES of Applicability of the LCO.	Yes	Yes	Yes	Yes
01-06 M	The new ACTION adds an explicit requirement to be in MODE 3 in 6 hours and MODE 4 in 12 hours if any SG loop has less than two MSSVs operable. This is one hour less than allowed by LCO 3.0.3.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-07 A	The CTS SR is revised to specifically reference the IST Program. The surveillance directly references Table 3.7-2 for lift points and incorporates the footnote from the table requiring the MSSV as left liftpoints to be within +/- 1% of the nominal setpoint.	Yes	Yes	Yes	Yes
01-08	Not used.	NA	NA	NA	NA
01-09 LS31	This DCPP specific change revises the maximum power range neutron high flux trip setpoints required for one or more inoperable MSSVs in accordance with the recommendations of Westinghouse NSAL 94-001, dated January 20, 1994, and specific analysis and calculations performed to confirm the conclusions of the Westinghouse NSAL.	Yes; LAR 97-06 submitted justifying revised high flux trip set points for inoperable MSSVs.	No; refer to 01-04-LS3.	No; refer to 01-04-LS3.	No; refer to 01-04-LS3.
01-10 LG	The Note on Table 3.7-2 stating that the set pressures shall correspond to the ambient conditions of the valve at normal operating temperatures is moved to the Bases of ITS SR 3.7.1.1.	Yes	Yes	Yes	Yes
01-11 LG	The MSSV line orifice size is moved to a licensee-controlled document.	Yes; moved to FSAR.	Yes; moved to FSAR.	Yes; moved to USAR.	Yes; moved to FSAR.
01-12 A	The proposed change would require that the plant be placed in HOT SHUTDOWN within 12 hours instead of COLD SHUTDOWN within 36 hours.	No; already part of CTS.	No; part of CTS.	Yes	Yes
02-01 LG	The descriptive material, definition of an AFW train, in the LCO is moved to the Bases.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-02 LS5	The ACTION specifies the requirements for AOT should one of the steam supplies to the TDAFW pump become inoperable.	Yes	No; part of CTS.	Yes	Yes
02-03 M	ACTIONS are modified to require restoration of the systems to meet the LCO within 10 days of discovery of failure to meet the LCO.	Yes	Yes	Yes	Yes
02-04 M	In this DCPD specific revision, the Applicability and ACTIONS are revised to include MODE 4 when the SGs are relied upon for heat removal.	Yes	No	No	No
02-05 A	The addition of the Note for three inoperable AFW trains assures that TS 3.0.3 will not be entered and that no other ACTION statement for other inoperable components will be applied that might force the unit into an unsafe condition.	Yes	Yes	Yes	Yes
02-06 LG	The change would move details regarding AFW motor operated discharge valves and ESW supply valve descriptions to the Bases.	No; descriptive material not part of CTS.	No; descriptive material not part of CTS.	Yes	Yes
02-07 M	The verification of valve alignments is revised to include automatic valves.	Yes	Yes	Yes	Yes
02-08 LS6	The surveillance interval for the AFW pump performance is changed from 92 days on a STB to "in accordance with the IST Program."	Yes	Yes	Yes	Yes
02-09 A	This DCPD specific surveillance is revised to combine the water and steam flow paths.	Yes	No	No	No

DCPD Conversion Comparison Table - Current TS

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-10 LS21	The time to achieve HOT SHUTDOWN if ACTIONS are not completed is changed from 6 to 12 hours. This time is reasonable to reach the required conditions under the circumstances since the SG heat removal system is the system performing the cooldown to MODE 4.	Yes; also refer to 02-04-M.	No; refer to 02-20-LS35.	No; refer to 02-20-LS35.	No; refer to 02-20-LS35.
02-11 A	In this DCPD specific revision, the testing requirements for pumps and valves are separated into two SRs.	Yes	No	No	No
02-12 TR1	The SR is revised to allow credit for an actual actuation and moves signal specifics to the Bases.	Yes	Yes	Yes	Yes
02-13 A	In this DCPD specific revision, the verification that the fire water storage tank is capable of realignment as an AFW water source is moved to the AFW ITS.	Yes	No	No	No
02-14 M	The Note for testing of the steam TDAFW pump is revised to explicitly define when testing must be performed.	Yes	Yes	Yes	No; part of CTS.
02-15 LG	The pump performance testing is revised to move the acceptance criteria to the Bases.	No; acceptance criteria not specified in CTS.	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-16 A	ACTION a. is clarified to note that ACTION [d.] conditions are not included.	No; CTS does not contain an Action for TDAFW pump supplies.	Yes	Yes	Yes
02-17 A	The CPSES specific note with respect to Unit 2 Train A is deleted since 2RFO2 is complete.	No	Yes	No	No
02-18 M	The surveillance is revised to be consistent with NUREG-1431 which requires verification of flow from the CST to each SG.	No; flow path verified at each startup.	No; SR not in CTS.	Yes	Yes
02-19 LG	The Callaway specific ACTIONS b and d are deleted since the LCO now refers to AFW trains versus pumps.	No	No	No	Yes
02-20 LS35	The time from failure to meet the ACTION requirements to shut the plant down would be revised from achieving HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours to achieving HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 12 hours.	No; refer to CN 02-10-LS21	Yes	Yes	Yes
02-21 LG	The Callaway specific requirement to perform the 18 month surveillance "during shutdown" would be moved to the Bases.	No	No	No	Yes
03-01 LG	The DCPD specific description of the AFW supply pathway and the required surveillances are moved to the AFW ITS and Bases.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-02 LS22	The requirement to verify the OPERABILITY of the ESW system when acting as backup to the CST for AFW supply by verifying the system is in operation is deleted. The surveillance will now require that the OPERABILITY of the backup system be verified "by administrative means."	No, the service water system is not a credited AFW backup supply	No, refer to 03-03-LG.	Yes	Yes
03-03 LG	The CPSES specific description of how to verify the OPERABILITY of the service water system when acting as backup to the CST for AFW supply is moved to the Bases. The surveillance will now require that the OPERABILITY of the backup system be verified "by administrative means."	No, the service water system is not a credited AFW backup supply	Yes	No, refer to 03-02-LS22.	No, refer to 03-02-LS22.
04-01 M	Isotopic analysis for DOSE EQUIVALENT I-131 concentration is to be performed on a 31 day frequency. The conditional performance requirements in the CTS are deleted.	Yes	Yes	Yes	Yes
04-02 LS8	The SR of Table 4.7-1 to determine gross radioactivity is deleted.	Yes	Yes	Yes	Yes
05-01 LS9	The MODE of Applicability for MSIVs is revised to clarify that in MODES 2 and 3 with all MSIVs closed [and de-activated], the safety function of the MSIVs is met.	Yes	Yes	No, maintaining CTS.	No, maintaining CTS.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-02 LS11	This DCPD specific change revises the AOT/completion time for an inoperable MSIV from four hours to eight hours in MODE 1 and in MODES 2 and 3. This change also deletes the MODE 1 requirement to place the plant in MODE 3, and then MODE 4 due to an inoperable MSIV and only requires entry into MODE 2 since the ACTIONS for MODE 2 or 3 operation would then be applicable. Operation in MODES 2 and 3 is revised to allow more than one MSIV to be inoperable and by a note to allow a separate condition entry for each inoperable MSIV.	Yes	No	No	No
05-03 LS12	The completion time for closing one inoperable MSIV is extended to 72 hours and separate required ACTIONS are included for either one MSIV inoperable or two or more MSIVs inoperable in MODES 2 or 3.	No, refer to 05-02-LS11.	Yes	Yes	Yes
05-04 A	The LCO is changed from the OPERABILITY of each MSIV to the OPERABILITY of four MSIVs.	Yes	No, already in CTS	Yes	Yes
05-05 M	A new requirement for a 7 day periodic verification of the closure [or isolation] of inoperable isolation valves is added.	Yes	No, already in CTS	Yes	Yes
05-06	Not used.	NA	NA	NA	NA
05-07 LS23	A footnote is added to the SR (in lieu of the current exception to TS 4.0.4) to indicate that demonstration of isolation valve operability is only required to be performed for entry into (and continued operation in) MODES 1 and 2.	Yes	Yes	No, part of CTS.	No, part of CTS.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-08 M	This change creates a new SR for the MSIVs and MFIVs to distinguish between the IST and the automatic actuation testing of these isolation valves. The surveillance allows credit for an actual actuation, if one occurs, to satisfy the SRs.	Yes	Yes	Yes	Yes
06-01 LG	This DCPP specific change revises the LCO to move the descriptive material to the Bases.	Yes	No	No	No
06-02 LS14	Adds an exception to LCO 3.0.4 for the 7 day action statement to restore the [ADV] operability.	Yes	Yes	No; part of CTS.	No; part of CTS.
06-03 M	Revises the specification to be applicable to the [ADV] lines, rather than only to the [ADV]. This includes the [block] valves.	No; CTS includes the block valves.	Yes	Yes	Yes
06-04 M	Surveillance is added that requires the manual cycling of the [block] valves [every 18 months.]	Yes	Yes	Yes	Yes
06-05 LS24	Adds a new ACTION for three or more inoperable [ADV] that requires action within 24 hours.	Yes	Yes	No; similar requirement in CTS.	No; similar requirement in CTS.
06-06 LG	Moves the requirements for the surveillances to the Bases and the testing specifics to licensee controlled documents.	Yes; testing specifics moved to FSAR.	Yes; testing specifics moved to TRM.	No; not in CTS.	No; not in CTS.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
06-07 LS25	The ACTION for [ADV] inoperable due to seat leakage is deleted. Seat leakage is no longer a condition of OPERABILITY.	No; CTS do not include seat leakage as operability criteria.	No; CTS do not include seat leakage as operability criteria.	Yes	Yes
06-08 M	Removes the exemption to TS 3.0.4 from the ACTION dealing with more than one inoperable [ADV].	No; TS 3.0.4 exemption not in CTS.	No; TS 3.0.4 exemption not in CTS.	Yes	Yes
06-09 A	This DCPD specific change revises the LCO to refer to the [ADV] lines versus [ADVs].	Yes	No	No	No
06-10 A	The CPSES specific action to place the RCS/RHR in operation is included in ITS 3.4.6.	No	Yes	No	No
07-01 A	The LCO and ACTIONS are changed from the OPERABILITY of each main feedwater line to the OPERABILITY of four [MFIVs and associated bypass valves] [and the requirement to have the valves closed or isolated if inoperable, is moved to APPLICABILITY].	Yes	Yes	Yes	Yes
07-02 LS37	The LCO Applicability is revised to exclude valves that are closed [and de-activated or isolated by a closed manual valve].	Yes	Yes	No, maintaining CTS.	No, maintaining CTS.
07-03 LS15	A Note is added to the ACTION statement that allows a separate condition entry for each inoperable valve.	Yes	Yes	No, refer to 07-16-LS34.	No, refer to 07-16-LS34.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
07-04 LS16	The DCPD specific AOT is revised for an individual inoperable valve from 4 hours to 72 hours. The ACTIONS are revised to add a new ACTION and eight hour AOT for two inoperable valves in the same flowpath to address the loss of isolation capability.	Yes	No	No	No
07-05 LS17	The CPSES specific action for inoperable feedwater isolation valves in MODE 1 is revised to include the option to isolate or close the inoperable valve and the restriction of the action to an inoperable "but open" valve is deleted. The ACTION for inoperable feedwater isolation valves in MODES 2 and 3 is revised to include the option to isolate the inoperable valve. The ACTIONS for inoperable feedwater isolation bypass valves in MODES 1, 2, and 3 are revised to include the option to isolate the inoperable valve.	No	Yes	No	No
07-06 LS33	This change revises the ACTION to apply to one "or more" feedwater valves inoperable.	Yes	Yes	Yes	Yes
07-07 A	The ACTIONS were revised that require achieving MODE 4 within 12 hours when the ACTIONS or Completion Times are not met.	No, part of CTS.	No, part of CTS.	Yes	Yes
07-08 LS26	The CPSES specific allowed Completion Time for an inoperable feedwater isolation valve is revised from 4 to 72 hours when credit for the feedwater control and associated bypass valves may be taken and is verified within four hours.	No, refer to 07-04-LS16	Yes	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
07-09 LS10	A new CPSES specific ACTION is added that retains the ACTION requirement which was modified by CN 07-08-LS26 and revises the Completion Time from four to eight hours.	No	Yes	No	No
07-10 M	A new CPSES specific surveillance for each feedwater control and associated bypass valve is added.	No, surveillance already in CTS	Yes	No	No
07-11 M	Verification of valve [isolation or] closure once every 7 days is added.	Yes	Yes	Yes	Yes
07-12 LS4	The ACTIONS associated with inoperable [MFIVs] would be revised to provide the alternative of closing [or isolating] an inoperable valve.	No, part of CTS.	No, refer to 07-05-LS17.	Yes	Yes
07-13	Not used.	NA	NA	NA	NA
07-14	Not used.	NA	NA	NA	NA
07-15	Not used.	NA	NA	NA	NA
07-16 LS34	CTS are revised to add a Note to allow separate condition entry for each inoperable [MFIV].	No, refer to change 07-03-LS15.	No, refer to change 07-03-LS15.	Yes	Yes
07-17 A	The CPSES specific allowance to open a valve closed per an ACTION in this specification is enveloped by ITS SR 3.0.5.	No	Yes	No	No
08-01 LG	The requirement to perform ACOTs and CHANNEL CALIBRATIONS would be moved to licensee controlled documents.	No, not in CTS.	No, not in CTS.	Yes, moved to USAR.	Yes, moved to FSAR.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
08-02 A	A Note is added to ACTION that references potential interaction with ITS 3.4.6 for RHR MODE 4 operability.	Yes	Yes	Yes	Yes
08-03 LG	The requirement to perform the 18 month surveillance "during shutdown" would be moved to the Bases.	No, not in CTS.	No, not in CTS.	Yes	Yes
08-04 A	A Note is added that clarifies [CCW] operability.	Yes	Yes	Yes	Yes
08-05 A	Surveillance is modified to exclude valves that are locked, sealed, or otherwise secured in their correct position.	Yes	Yes	Yes	Yes
08-06 TR1	The SR is revised to allow credit for an actual actuation and moves signal specifics to the Bases.	Yes	Yes	Yes	Yes
08-07 M	A new surveillance specific to DCPD is added that requires verifying that each CCW pump starts automatically on an actual or simulated signal actuation at least once per 18 months.	Yes	No	No	No
08-08 A	Surveillance is modified to only be applicable to flow path valves.	Yes	Yes	Yes	Yes
08-09	Not Used.	NA	NA	NA	NA
08-10 A	The Callaway specific Note applicable to cycle 1 surveillance requirements is no longer needed.	No	No	No	Yes
09-01 M	A DCPD existing surveillance is revised that requires verifying that a motive source is available that would allow the required valves to be repositioned.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-02 A	A Note is added that requires entry into applicable LCOs if an inoperable [ASW] system makes the affected equipment inoperable.	No, ASW only supplies CCW heat exchangers.	Yes	Yes	Yes
09-03 A	The DCPD specific surveillance is reworded and the requirement to verify the position of automatic valves is deleted since the ASW system has no automatic valves.	Yes	No	No	No
09-04 M	A new surveillance specific to DCPD is added that requires verifying that each ASW pump starts automatically on an actual or simulated actuation signal at least once per 18 months.	Yes	No	No	No
09-05 TR1	The SR is revised to allow credit for an actual actuation and moves signal specifics to the Bases.	No, refer to 09-04-M. This requirement did not previously exist.	Yes	Yes	Yes
09-06 M	A new surveillance specific to DCPD is added that requires verifying that each valve in the flow path that is not locked, sealed, or otherwise secured in position is capable of being placed in the correct position in accordance with the IST Program.	Yes, also refer to change 09-01-M.	No	No	No
09-07 A	A Note is added to the [ASW] surveillance that clarifies system OPERABILITY requirements. Isolation of [ASW] flow to individual components does not render the system inoperable.	No, ASW only supplies CCW heat exchangers.	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
10-01 LG	The DCPD specific text description, definition of a ventilation train, is deleted from the LCO and moved to the Bases.	Yes	No	No	No
10-02 M	The Applicability and ACTIONS are revised to include "during movement of irradiated fuel assemblies."	Yes	No, part of CTS.	Yes	Yes
10-03 LS7	The SR for control room ventilation system is revised to require the filtration units without electric heaters to be tested for only 15 minutes instead of 10 hours.	No, plant configuration includes heaters.	Yes	No, refer to 10-22-M.	No, refer to 10-22-M.
10-04 A	An ACTION statement is added to require entering 3.0.3 if two trains of the control room (CR) ventilation filter system are inoperable in MODES 1, 2, 3, or 4.	Yes	Yes	Yes	Yes
10-05 LS18	A new option is added to the ACTIONS by NUREG 1431 that allows the suspension of CORE ALTERATIONS or movement of irradiated fuel versus placing the ventilation system in the recirculation mode.	Yes	No, part of CTS.	Yes	Yes
10-06 LG	The details and description of the Required Actions and the monthly SRs for train operability are moved to the Bases.	Yes	No, not in CTS.	Yes	Yes
10-07 LG	The surveillance that verifies CR temperature once per 12 hours is moved to a licensee-controlled document.	Yes, moved to ECGs.	No, not in CTS.	Yes, moved to USAR.	Yes, moved to FSAR.

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
10-08 A	The description of the ventilation filter specific testing requirements are moved to the VFTP, as defined in the Administrative Controls of the ITS, or deleted as being duplicated in the applicable RGs or Standards. A SR is added that requires [CR and Auxiliary Building ventilation system] filter testing in accordance with the VFTP.	Yes	Yes	Yes	Yes
10-09 LS27	The ACTION for an OPERABLE ventilation train not being capable of being supplied from an emergency power source is deleted.	No, refer to change 10-16-LG.	No, not in CTS.	Yes	Yes
10-10 TR1	The SR is revised to allow credit for an actual actuation and moves signal specifics to the Bases.	Yes	Yes	Yes	Yes
10-11 LS19	Frequency of the surveillance requiring verification of the control room ventilation system capability to maintain a positive pressure in the CR is relaxed to 18 months on a STB.	Yes	Yes	Yes	Yes
10-12 LS32	Deletes the STB for the 31 day testing.	No, CTS surveillance is not STB.	No, not in CTS.	Yes	Yes
10-13 LG	The DCPD specific footnotes indicating the control room ventilation system is common to both units and that the system may be considered OPERABLE with no chlorine monitors if no bulk chlorine gas is stored within the SITE BOUNDARY, are moved to the Bases.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
10-14 A	The statement that LCO 3.0.4 is not applicable is deleted based upon the new ITS definition of LCO 3.0.4 which does not apply in MODES 5 and 6.	No, TS 3.0.4 exemption is not in CTS.	No, TS 3.0.4 exemption is not in CTS.	Yes	Yes
10-15 LG	The ventilation system flow rates would be moved to licensee controlled documents. These flow rates are established in conjunction with flow balancing of the ventilation systems.	No, retained in CTS.	No, retained in CTS.	Yes, to USAR.	Yes, to FSAR.
10-16 LG	The DCPD specific requirement to have an available emergency power source for the control room ventilation system is moved to the Bases. This change is consistent with the Bases of NUREG-1431 for ventilation systems required during fuel movement.	Yes	No	No	No
10-17 A	The SR to measure ventilation system flow rate is not identified as a separate SR in the ITS because it is verified during the other in place filter tests (see ITS 5.5.11).	Yes	Yes	Yes	Yes
10-18 LS36	The CPSES specific action shutdown requirement is revised from MODE 3 in the next 6 hours and MODE 5 in the following 30 hours to enter LCO 3.0.3. This effectively adds up to one hour to the completion time.	No	Yes	No	No
10-19 A	Clarifies that for CPSES, the "pressurization" mode is called the "emergency recirculation" mode.	No	Yes	No	No
10-20 LS39	This change establishes appropriate Required Actions and Completion Times for ventilation system pressure envelope degradation.	No	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
10-21 LS38	The ACTION to immediately suspend positive reactivity changes when both trains of control room ventilation are inoperable in MODES 5 and 6 and during movement of irradiated fuel assemblies is deleted.	Yes	No, not in CTS.	Yes	Yes
10-22 M	A requirement to operate the filtration system fans for at least 15 minutes would be added consistent with NUREG-1431.	No, the CTS requires 10 hours.	No, refer to 10-03-LS7.	Yes	Yes
10-23 LS13	The "within 31 days after removal" requirement for completion of laboratory analyses is deleted. This requirement is not contained in the ITS nor is it contained in RG 1.52 or the applicable ANSI standards.	Yes	Yes	Yes	No, retaining CTS.
10-24 A	The 30°C temperature specified for laboratory testing of filter carbon samples is added. [This temperature is specified by ASTM D 3803-1989, but is also explicitly stated in NUREG-1431.]	Yes	Yes	No, part of CTS.	No, part of CTS.
11-01 M	A new specification for CR heat removal is added to ensure that the CR equipment functions following a DBA.	No, not part of CTS.	No, part of CTS.	Yes	Yes
11-02 LS28	Extends the AOT for an inoperable CR A/C unit from 7 to 30 days.	No, not part of CTS.	No, part of CTS.	Yes	Yes
12-01-LS29	The Frequency of the surveillance for verification of the ESF filtration train capability to maintain a negative pressure relative to atmospheric pressure is revised from 18 months to 18 months on a STB.	No, CTS does not require the surveillance.	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-02 M	A new ACTION is added to cover the condition when both emergency exhaust system trains are inoperable in MODES 1, 2, 3, or 4.	No, addressed by ITS LCO 3.0.3.	No, addressed by ITS LCO 3.0.3.	Yes	Yes
12-03	Not Used.	N/A	N/A	N/A	N/A
12-04 TR1	The SR is revised to allow credit for an actual actuation and moves signal specifics to the Bases.	Yes	Yes	Yes	Yes
12-05 LS32	Deletes the STB for the 31 day testing.	No, STB not part of CTS.	Yes	No, refer to 3/4.9 SRs.	Yes
12-06 LG	The details and description of the SR for train OPERABILITY are moved to the Bases.	Yes	Yes	No, not in CTS.	No, not in CTS.
12-07 M	A new CPSES specific surveillance is added to verify that non safety-related fans stop upon initiation of an actual or simulated signal.	No	Yes	No	No
12-08 M	A new Callaway specific surveillance is added to verify the capability of the emergency exhaust system (EES) to maintain a negative pressure. A Note is also added to the Applicability of the EES TS to clarify when both EES functions are required to be OPERABLE.	No	No	No	Yes
12-09 A	The Callaway specific surveillance is revised to specify the ventilation system line-up that is required for performance of the negative pressure test.	No	No	No	Yes
12-10 M	The DCPD specific CTS ABVS LCO is extended to the supply fans.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
13-01 LG	The DCPD specific Note stating that the UHS is common to both units is moved to the Bases.	Yes	No	No	No
13-02 LG	The Callaway specific descriptive material is moved to the Bases.	No	No	No	Yes
13-03 R	The CPSES specific LCO for sediment depth and the surveillances for SSI Dam inspections and sediment depth are relocated to a licensee-controlled document.	No	Yes, relocated to the TRM.	No	No
13-04 LG	This Wolf Creek specific change moves the description of the required dam height and the requirement for the related surveillance to a licensee controlled document.	No	No	Yes, moved to the USAR.	No
13-05 LS30	The CPSES specific ACTION associated with the restoration of SSI level is revised to allow seven days for completion of the required action.	No	Yes	No	No
13-06 LG	This Callaway specific change moves the 18 month visual inspection of the cooling tower fill material to a licensee controlled document.	No	No	No	Yes, moved to the FSAR.
13-07 LG	This Callaway specific change moves the 31 day UHS riprap inspection to a licensee-controlled document.	No	No	No	Yes, moved to the FSAR.
13-08 M	This DCPD specific change extends the LCO Applicability to MODE 4 and revises ACTIONS accordingly.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
13-09 A	The DCPD specific exemption to 3.0.4 is deleted. Placing the second CCW heat exchanger in service per the ACTION allows operation to continue indefinitely, thus per NUREG-1431 LCO 3.0.4 is not applicable.	Yes	No	No	No
14-01 R	The CPSES specific Steam Generator Pressure/Temperature Limitation TS is relocated to a licensee-controlled document.	No	Yes, relocated to the TRM.	No	No
15-01 R	The CPSES specific Flood Protection TS is relocated to a licensee-controlled document.	No	Yes, relocated to the TRM.	No	No
16-01 R	The CPSES specific Snubbers TS is relocated to a licensee-controlled document.	No	Yes, relocated to the TRM.	No	No
17-01 R	The CPSES specific Area Temperature Monitoring TS is relocated to a licensee-controlled document.	No	Yes, relocated to the TRM.	No	No
18-01 A	A clarifying CPSES specific Note is added concerning isolation of the safety chilled water flow to individual components to mimic the CCW TS which is similar to this TS.	No	Yes	No	No
18-02 LG	The CPSES specific surveillance for the electrical switch gear area emergency fan coil units is moved to a licensee-controlled document.	No	Yes, moved to the TRM.	No	No
18-03 TR1	The CPSES specific SR is revised to allow credit for an actual actuation and moves signal specifics to the Bases.	No	Yes	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
19-01 LG	The CPSES specific Main Feedwater Pressure/Temperature limit TS is relocated to a licensee-controlled document.	No	Yes, relocated to the TRM.	No	No
20-01 R	Moves what constitutes an OPERABLE CPSES specific UPS train to the Bases.	No	Yes	No	No
20-02 TR1	The CPSES specific UPS activation surveillance is revised to allow credit for an actual actuation and moves signal specifics to the Bases.	No	Yes	No	No

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION

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	LS26.....	Not applicable to DCP
	LS27.....	Not applicable to DCP
	LS28.....	Not applicable to DCP
	LS29.....	Not applicable to DCP
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NO SIGNIFICANT HAZARDS CONSIDERATION

(continued)

IV. Specific No Significant Hazards Considerations - "LS" (continued)

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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

DCPP No Significant Hazards Evaluations

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS.

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3..." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"

10CFR50.92 EVALUATION FOR ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

DCPP No Significant Hazards Evaluations

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

DCPP No Significant Hazards Evaluations

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications.

DCPP No Significant Hazards Evaluations

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION
BASES, FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

DCPP No Significant Hazards Evaluations

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?
The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident

DCPP No Significant Hazards Evaluations

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1

10 CFR 50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A Note is added to allow separate condition entry and completion times for each inoperable main steam safety valve (MSSV). In the event of an inoperable MSSV, the CTS require that any MSSVs discovered to be inoperable, be returned to an OPERABLE condition or that the power range neutron flux high trip setpoint be appropriately reduced within the four hours provided. The most probable cause for an inoperable MSSV is the failure to meet the surveillance required as-found lift pressure. The intent of the four hours is to allow time to reset the valve lift point to within required tolerances or reduce power and the power range neutron flux high trip setpoints. The NUREG-1431 ACTION statement and the CTS inferred ACTION for the required power reduction is based on an inoperable MSSV per steam generator (SG).

The CTS requires that all MSSVs discovered to be inoperable be returned to an OPERABLE status within the four hours provided for the first inoperable valve. For example, if a second inoperable MSSV was discovered three hours into the ACTION statement of the first inoperable MSSV, both MSSVs would have to be returned to OPERABLE within the following one hour or the unit power level would have to be reduced in accordance with the CTS. The addition of the Note allowing separate entry for each inoperable MSSV provides the full four hours for each inoperable MSSV. This is acceptable based upon the assumptions used to analyze the unit with inoperable MSSVs (highest flow MSSV fails to open and the other three MSSVs with similar lift points on the other SGs also fail).

The accident scenarios of concern involve a loss of load (LOL) or turbine trip (TT) without a direct, or with a delayed, reactor trip. However, although it is not credited in the accident analysis (because the analysis was meant to evaluate a LOL as well as a turbine trip), the reactor protection system will cause a reactor trip on a turbine trip. This would avoid excessive heat input into the secondary side and provide further protection to prevent overpressurization.

The lift pressures of the MSSVs were determined assuming that the [atmospheric steam dump] valve [(ADV)] on each steam line was non-safety class and conservatively were not assumed to function in the licensing basis analyses. However, the [ADVs] themselves are safety grade and are expected to function to provide overpressure control [even though the automatic pressure control portion of the control circuitry is not Design Class I. However, the control circuitry for manual operation is Design Class I; thus manual operation of the ADVs would be able to relieve pressure in the long term.] The non safety grade condenser steam dumps would also be expected to provide overpressure control in the event of a turbine trip. With a loss of offsite power, the condenser steam dumps will not function due to the loss of condenser vacuum.

[The inoperability of the lowest set MSSV can potentially affect auxiliary feedwater (AFW) flow to the SGs following the above transients. However, if the lowest set MSSV is inoperable the lift point of the next higher set MSSV establishes the back pressure that the AFW pump must pump against. Adequate AFW flow has been verified as being available.]

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (continued)

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or changes to operating methodologies. This revision does not affect an accident initiator of any analyzed accident. This change is bounded by the analysis performed for an inoperable MSSV on each SG, which assumed that all of the valves at the lowest lift point fail to open. Thus, the multiple conditional entry for one valve inoperable on each SG is enveloped. The allowed multiple condition entry for inoperable valves on the same SG is predicated on the low probability of an event that would require the MSSVs to function and recognizes the availability of the other OPERABLE MSSVs, the [ADV(s)], the reactor trip from turbine trip, and the availability of the condenser steam dumps for transient pressure mitigation.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function. The existing analysis for inoperable MSSVs envelopes the multiple entries since it assumes that one MSSV on each SG is inoperable. The allowed multiple condition entry for inoperable valves on the same SG is predicated on the low probability of an event that would require the MSSVs to function and recognizes the availability of the other OPERABLE MSSVs, the [ADVs], the reactor trip from turbine trip, and the availability of the condenser steam dumps for transient pressure mitigation.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (continued)

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not change any accident analysis assumptions, initial conditions, or results. Consequently, it does not have an effect on a margin of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS1" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS allow continued operation with inoperable MSSVs if the power range neutron flux high reactor trip setpoints are reduced. The amount of reduction in the trip setpoint is dependent on the total number of inoperable MSSVs per SG and is intended to compensate for the lost relief capacity (heat removal capability and thus overpressure protection) should a transient requiring their operation occur. In the proposed specification, the CTS requirement to reduce the power range high neutron flux reactor trip setpoint is retained; however, the time to complete resetting the trip setpoints would be changed from four to 72 hours.

The CTS require that, if the MSSV cannot be restored to an OPERABLE status within four hours, the power range high neutron flux reactor trip setpoints must be reset in the same 4-hour period. NUREG-1431 requires that the reactor power be reduced in four hours if the MSSV cannot be returned to an OPERABLE status; however, NUREG-1431 would not require resetting the power range neutron flux high setpoints. The Westinghouse Owners Group (WOG) has proposed changes to NUREG-1431 (traveler WOG-83, as revised through draft Rev. 1) that: 1) propose that the completion time for resetting the power range neutron flux high trip setpoint to compensate for a positive MTC or a control rod withdrawal event at partial reactor power to be 72 hours, 2) specifies that power level reductions be per the Westinghouse Nuclear Safety Advisory Letter, NSAL 94-01 and 3) deletes the Maximum Allowable % RTP for 5 MSSVs OPERABLE. However, pending approval of draft Rev. 1 of WOG-83, the changes proposed in the traveler have been modified to retain the current TS requirement to reset the power range neutron flux-high trip setpoints based on the number of MSSVs inoperable to a maximum allowable power determined in accordance with calculations or analysis to account for Westinghouse NSAL 94-001. The allowed Completion Time to reduce the Power Range Neutron Flux trip setpoints is reasonable based on operating experience to accomplish the required ACTIONS in an orderly manner. The power levels specified per NSAL 94-001 are based on a conservative algorithm developed by Westinghouse to bound the required relief capacity.

The above changes are consistent with NUREG-1431 as revised by WOG-83 and NSAL-94-001. This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3 (continued)

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or changes to operating methodologies. This revision does not affect an accident initiator of any analyzed accident. The proposed change is based on recognition that: 1) Reactor power must be limited to compensate for lost pressure relief (heat removal capability) with inoperable MSSVs; and 2) that there is a need to prevent a reactor power increase via a reduced power range neutron flux high trip setting, for transients that would cause added heat transfer to the secondary system. The proposed change also allows for an appropriate amount of time to: 1) restore the MSSV to OPERABLE or reduce power; and 2) reduce the power range neutron flux high reactor trip setpoints. The probability of an accident occurring and requiring MSSV operation during the 72-hour Completion Time would be very small. The time extension allows the reactor trip setpoint reduction to be performed in a more deliberate manner, thereby reducing the potential for inadvertent reactor trips introduced during the setpoint reduction. However, because the accident analyses are unaffected by the proposed change, the consequences of the accident analyses are not affected by this change.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not require physical alteration to any plant system or change the method by which any safety-related system performs its function. No changes in plant operation result from the changes, and no new event initiators are introduced.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not change any accident analysis assumptions, initial conditions or results. The required reduction in power level and the required reduction in power range neutron flux high reactor trip setpoints both restore the assumed margin of safety. The extension of time to reset the reactor trip setpoints does not have a significant effect on a margin of safety because the likelihood of the initial conditions and the occurrence of the sequence of events that would be required to challenge the MSSVs during the time period is very small. In addition, the Completion Time for reducing reactor power remains unchanged. Furthermore, the accident analyses are assumed to be initiated from conditions which are consistent with the TS LCOs. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Consequently, this change does not have a significant effect on a margin of safety.

DCPP No Significant Hazards Evaluations

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

**NSHC LS3
(continued)**

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-3" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS define the steam turbine-driven (TDAFW) pump as OPERABLE if it has two OPERABLE and redundant steam supplies. Since no ACTIONS are specified for an inoperable steam supply, the TDAFW pump must be declared inoperable if one of the two redundant steam supplies becomes inoperable. The requirement for two supplies is based upon the SG tube rupture (SGTR) and main steam line break (MSLB) accidents to assure that a 100 percent redundant steam supply to the TDAFW pump is available.

NUREG-1431 provides a Required Action if one of the steam supplies becomes inoperable, thus avoiding inoperability of the pump. NUREG-1431 recognizes that the TDAFW pump can continue to perform its intended function with only one steam supply since the supplies are redundant. NUREG-1431 requires that with only a single steam supply OPERABLE that a Required Action be entered. The time to return the inoperable steam supply to an OPERABLE condition is seven days. This time is reasonable considering that: 1) there is an OPERABLE 100 percent redundant steam supply; 2) there are two OPERABLE motor-driven AFW pumps; and 3) there is a low probability of an SGTR or MSLB event occurring that would require the use of the inoperable steam supply.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or changes to operating methodologies. This revision does not affect an accident initiator of any analyzed accident since the TDAFW pump steam supply only provides power to equipment required to mitigate the consequences of an accident.

The revision recognizes the low probability of an accident requiring the use of the inoperable steam supply for the TDAFW pump coincident with the failure of the motor-driven AFW pumps.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5 (continued)

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not change any accident analysis assumptions, initial conditions, or results. Consequently, it does not have an effect on margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS5" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The surveillance interval for the AFW pump performance test is revised from a Frequency of 92 days on a STAGGERED TEST BASIS to the Frequency specified by the IST Program. This change will result in all TS pump performance surveillance tests referencing the IST Program for frequency and test requirements. This change will eliminate any potential ambiguity associated with AFW pump testing as a result of ASME changes and results in consistent presentation of pump testing throughout the TS. The inservice tests confirm component operability, trend performance, and detect incipient failures by indicating abnormal performance. This change is classified as a relaxation since the frequency is no longer specified in the TS. The pump testing frequency will now be controlled by the IST Program which may be changed within the limits of the ASME code without prior NRC written approval per Generic Letter (GL) 89-04, Supplement 1. This change has been proposed as a change to NUREG-1431 via TSTF-101.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not alter the unit configuration or operation or the function of any safety system. Consequently, the changes do not increase the probability of an accident as defined in the FSAR Update. Revising the testing frequency or the reference for the testing frequency for an accident mitigating system consistent with NRC guidance does not effect the analyzed accident, its probability, or its consequences.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system is tested or performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements or change any accident analysis assumptions, initial conditions or results. Consequently, the proposed change does not involve a significant reduction in margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS6" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS require that the gross radioactivity of the secondary system coolant be determined every 72 hours. This surveillance is basically only an indicator of the potential offsite whole body dose. Since the radioiodines and the resulting thyroid dose are limiting, not noble gases and whole body dose, NUREG-1431 has deleted the 72 hour gross radioactivity surveillance requirement. Since the limits on primary to secondary leakage and DOSE EQUIVALENT I-131 assure that the dose analyses in the FSAR remain valid, the revised surveillance is more appropriate. NUREG-1431 also requires that the surveillance for verification of I-131 activity be performed every 31 days on an unconditional basis, which is more restrictive than the CTS.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or changes to operating methodologies. This revision does not affect an accident initiator of any analyzed accident. This change only deletes gross radioactivity sampling where results are bounded by the primary to secondary leakage and DOSE EQUIVALENT I-131 limits. The consequences of secondary system releases are limited by radioiodines and their resultant thyroid exposures, not the whole-body exposures received from the noble gases. In addition, the surveillance Frequency for verification of I-131 activity has been increased to 31 days unconditionally, which is more restrictive than the CTS.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

DCPP No Significant Hazards Evaluations

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8 (continued)

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function. The change only deletes a surveillance function utilized for trending.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not change any accident analysis assumptions, initial conditions, or results. Consequently, it does not have an effect on margins of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS8" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS require that if one main steam isolation valve (MSIV) is inoperable in MODES 2 or 3, operation may continue if that valve is maintained closed. The CTS do not recognize additional MSIVs being closed as meeting the LCO. Thus, if multiple MSIVs are closed, the unit is required to enter the ACTION statement.

NUREG-1431 recognizes that if all MSIVs are closed [and de-activated] and verified closed, their safety function is being fulfilled and there is no need to enter the ACTION. If the valves cannot be closed, then NUREG-1431 requires that the unit be taken to MODE 4 in 12 hours, which is consistent with the CTS ACTION for one MSIV that cannot be closed. Going to MODE 4 places the unit in a condition such that the LCO is no longer applicable.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or changes to operating methodologies. This revision does not affect an accident initiator of any analyzed accident. This change recognizes that since the MSIVs are closed, they are performing their intended function. Consequently, there is no need to place the unit in a MODE where the LCO is not applicable.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9 (continued)

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function. The MSIVs would be in their accident mitigating position, thus fulfilling their required function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not change any accident analysis assumptions, initial conditions, or results. Consequently, it does not have an effect on margins of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS9" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS11
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS ACTION statement requires that with one MSIV inoperable and open in MODE 1, OPERABILITY be restored within four hours, or the unit be placed in at least HOT STANDBY within the next six hours and in HOT SHUTDOWN within the following six hours. The CTS MODES 2 and 3 ACTION statement specifies that subsequent operation in MODE 2 or MODE 3 may proceed provided the one inoperable MSIV is closed and maintained closed. The CTS and NUREG-1431, LCO 3.0.3, state that where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the LCO. This logic can also be applied to specific ACTIONS requiring MODE transitions. In this case, if an MSIV could not be returned to an OPERABLE condition in four hours, the unit would be required to transition to MODE 4 as long as the MSIV remained open. If during the transition, the inoperable valve was closed, the ACTION requirements for operation in MODES 2 or 3 would be met and the transition to MODE 4 would not be required. Some repairs to an MSIV can be made during operation in MODE 1; however, it may not be possible to restore OPERABILITY within the AOT of four hours.

NUREG-1431 increases the AOT from four hours to eight hours for an inoperable and open MSIV. This revised AOT provides a more reasonable time to diagnose the problem, mobilize corrective ACTIONS, obtain administrative clearances, complete the maintenance, restore the valve to an OPERABLE condition, and where appropriate, perform a post-maintenance verification. The additional time reduces the probability of unnecessary unit transients and unit shutdowns, thus improving unit safety and increasing unit availability.

The CTS MODE 1 requirement to take the unit to HOT SHUTDOWN if the valve cannot be closed within four hours is overly restrictive. Once the transition to MODE 2 occurs, the ACTIONS of MODE 2 should apply. The ACTIONS of MODE 2 require the inoperable valve to be closed in 4 hours or the unit must be taken to HOT SHUTDOWN. To eliminate potential confusion, NUREG-1431 revises the MODE 1 ACTION to clarify the stages of unit shutdown required for an inoperable MSIV. The ACTION is revised to specify that if an inoperable MSIV is not restored to OPERABLE status within the proposed 8 hour AOT, the unit must be in MODE 2 within the next 6 hours and the MODE 2 requirements would apply. The Completion Time can be greater for these valves as opposed to other containment isolation valves because these valves are isolating a closed system that penetrates containment.

The safety function of the MSIVs is to close. Therefore, a closed MSIV will perform its safety function even if it is not OPERABLE. However, an MSIV cannot be closed with the unit operating in MODE 1 without initiating a unit shutdown. To provide the option to close the MSIV, the unit must be in at least MODE 2. The CTS MODES 2 and 3 ACTION specifies that subsequent operation with an inoperable MSIV may proceed provided that the MSIV is maintained closed. However, the frequency at which the valve is to be verified "maintained closed" is not specified, nor is the AOT to return the valve to an OPERABLE condition or the time to close an inoperable MSIV. NUREG-1431 recognizes that if the MSIV is closed, it is performing its safety function and that if other MSIVs become inoperable their closure will allow them to perform their safety function. The addition of the note allowing separate entry for each inoperable MSIV provides the full AOT for each inoperable MSIV. This is acceptable based upon the fact that the ACTION is to close the valve and also based on the low probability of a condition requiring the closure of the MSIVs in MODES 2 and 3 while in the proposed eight hour ACTION.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS11 (continued)

NUREG-1431 revises the MODES 2 and 3 ACTION to allow up to eight hours to restore the valve to an OPERABLE condition or close an inoperable MSIV. Additionally, NUREG-1431 MODES 2 and 3 ACTION requires an inoperable and closed MSIV to be verified closed at least once every 31 days. Verifying the position of a closed MSIV every 31 days is reasonable because MSIV status indications are available in the CR to verify the valve is closed and this frequency is consistent with the CIV STS 3.6.3. From a containment isolation consideration, these valves can have a longer completion time since the valves are isolating a closed system that penetrates containment.

The proposed changes to increase the MSIV AOTs are consistent with the recommendations from NUREG-1024, "Technical Specifications - Enhancing the Safety Impact." NUREG-1024 states in part:

Allowable outage times that are too short will subject the unit to unnecessary trips, transients and fatigue cycling. Outage times that are too short also may result in less thorough repair and post repair testing before equipment is returned to service.

Safety Function of MSIVs

The MSIVs must be open during power operation. The valves have a safety function to close to mitigate the consequences for the following:

1. The steam discharged into containment from a MSLB inside containment is limited by MSIV closure. After MSIV closure, only steam from the affected SG and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops is discharged into containment. For this accident scenario, steam is discharged into containment from all SGs until the unaffected are isolated by closure of their respective MSIVs. Closure of the MSIVs for the unaffected (SG) isolates them from the break.
2. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. However, the uncontrolled blowdown of more than one SG loop must be prevented to limit the extent of uncontrolled RCS cooldown and core positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single SG loop in the same manner as a break within the containment.
3. A break downstream of the MSIVs will be isolated by the closure of the MSIVs. A single failure of one MSIV will not result in uncontrolled blowdown of more than one SG loop.
4. Following a SGTR, closure of the MSIV for the affected SG isolates that SG from the intact SGs. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the SG with the ruptured tube independent of the other SGs, a necessary step toward equalizing the pressure difference across the tube rupture and terminating flow through the ruptured tube.
5. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting than a steam line break for MSIV operability, since the uncontrolled blowdown transient and resulting containment pressurization associated with a feedwater break is slightly less severe.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS11
(continued)

Summary and Conclusions

The effect on unit safety of an increase in the MSIV MODE 1 AOT from four hours to eight hours, the addition of a MODE 2 and 3 eight hour AOT to restore or close inoperable valves, and the allowance of multiple condition entries has been evaluated. The increase in AOT does result in a slight increase in risk; however, the risk associated with a unit shutdown due to the shorter AOT is reduced. In conclusion, there is reasonable assurance the health and safety of the public will not be adversely affected by the proposed TS changes.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change to the MSIV TS does not alter the unit configuration or operating methodologies or the function of any safety-related systems. Even though there is a slight risk increase associated with an increase in AOT, the risk is small and not significant. The proposed changes do not significantly increase the probability of an accident as defined in the FSAR Update.

Closure of inoperable MSIVs in MODES 2 or 3 and the allowed multiple condition entries that would eventually result in their closure places them in the position that fulfills their safety function. Therefore, continued unit operation in MODES 2 or 3 with inoperable MSIVs in separate condition entries or closed does not effect the consequences of an accident.

The 31 day surveillance frequency to verify the closure of an isolated valve is reasonable considering there is valve position indication in the CR, and since the valve is in its safeguard position, there are no safety concerns.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

DCPP No Significant Hazards Evaluations

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

LS11 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function. Closure of inoperable MSIVs places them in the position that fulfills their safety function, and consequently is not a physical alteration to the unit.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements or change any accident analysis assumptions, initial conditions, or results. Consequently, there is no significant effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS11" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS13
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

To assure charcoal adsorber OPERABILITY, the SR requires that a laboratory analysis be performed and the results obtained within 31 days of removing the charcoal sample. The sample must be sent to an offsite laboratory for this analysis. It is proposed that the time requirement of "within 31 days after removal" for completion of laboratory analyses be deleted. This requirement is intended to avoid extended plant operation with degraded charcoal filters. This requirement is not contained in the ITS nor is it contained in the RG 1.52 or the applicable ANSI standards. There is no safety significant basis for maintaining this time limit as a TS requirement. Laboratory analyses are performed under contract with a laboratory on a prompt basis, and it is not necessary to prescribe a time limit within TS for completing the analysis. Failure to complete an analysis within 31 days has insignificant safety consequences because the results would be available within approximately the same time period and it is very unlikely that the charcoal would be degraded to the extent that there would be a complete loss of a safety function.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or changes to operating methodologies. This revision does not affect an accident initiator of any analyzed accident. The revision recognizes that the timely completion of this surveillance will not affect the probability or consequences of any accident.

In addition, the results of charcoal filter testing would be available on approximately the same schedule as before. Thus, the proposed change would not have a significant effect on the availability of the filters to perform as assumed following an accident.

DCPP No Significant Hazards Evaluations

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

LS13 (continued)

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any plant system or change the method by which any safety-related system performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change would continue to assure that the charcoal filters perform as required although the time period for obtaining test results would be removed from TS. However, the results would be available on approximately the same schedule as before. Consequently, it does not have significant effect on a margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS13" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14
10 CFR 50.92 EVALUATION
FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

An exemption to TS 3.0.4 is added that allows MODE transition to occur with a single inoperable [ADV]. The CTS does not permit a MODE transition with an inoperable valve. The valve must be returned to an OPERABLE condition within seven days of becoming inoperable before the MODE transition is permitted. The CTS allow an [ADV] to be inoperable in MODES 1, 2, or 3, because the risk of an accident requiring [ADV] valve operation has been evaluated and found to be low. Allowing MODE transition with an inoperable [ADV] does not significantly increase that risk since the remaining valves are operable.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or changes to operating methodologies or affect the function of any safety-related system. This revision does not affect an accident initiator of any analyzed accident. This change recognizes that the probability of an accident requiring the single inoperable valve to function is low and is unlikely to occur during the AOT even with the permitted MODE transition.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

DCPP No Significant Hazards Evaluations

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14
(continued)

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements or change any accident analysis assumptions, initial conditions, or results. Consequently, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS14" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS15
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS are revised to apply to all four feedwater lines as a group. A Note is added to the CTS to allow separate condition entry time for each inoperable valve. NUREG-1431 recognizes that with one valve inoperable and open in the flow path, that feedwater isolation, within the assumptions used for accident analysis, is still functional via the safety grade MFIVs, [the safety grade main feedwater (MFW) regulating and bypass valves (that are located in non safety grade piping), or via the MFW pump non safety grade pump trip.]

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or alter the function of any safety-related system. This revision does not affect an accident initiator of any analyzed accident. This change recognizes that the probability of an accident requiring isolation of the flow is low and recognizes the availability of the feedwater pump trip upon a feedwater isolation signal to maintain the consequences of the assumed accident within the analyzed conditions.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS15 (continued)

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements or change any accident analysis assumptions, initial conditions, or results. Consequently, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS15" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS EVALUATION

NSHC LS16
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

With one feedwater isolation/regulating/bypass valve inoperable and open in MODE 1, 2, or 3, the CTS ACTION statement requires that OPERABILITY be restored within four hours. If OPERABILITY cannot be restored or valve closure accomplished, the MFIV must be closed or for the main feedwater regulating (MFRV)/bypass valves, the valve must be closed or isolated, or the unit must be placed in at least HOT STANDBY within the next six hours and in HOT SHUTDOWN within the following six hours. The CTS applies to each feedwater line individually. Some repairs to a valve can be made during operation in MODE 1, 2, or 3; however, it may not be possible to restore OPERABILITY within the AOT of four hours.

NUREG-1431 increases the MODE 1, 2, and 3 AOT from 4 hours to 72 hours for an inoperable and open valve. The increased AOT provides a more reasonable time to diagnose the problem, mobilize corrective ACTIONS, obtain administrative clearances, complete the maintenance, restore the valve to an OPERABLE condition, and where appropriate, perform a post-maintenance verification. The additional time reduces the probability of unnecessary unit transients and unit shutdowns, thus improving unit safety. NUREG-1431 recognizes that the inoperability of one valve in the flow path does not render the feedwater isolation function inoperable. This conclusion is based on the isolation redundancy provided by the MFIV and MFRV and associated bypass valves that are assumed to close in less than or equal to seven seconds by NUREG-1431.

An argument and a similar conclusion can be made for MFIVs that have closure times greater than that of the MFRVs and associated bypass valves. In addition, a containment pressure analysis has been performed for the large secondary line break with an assumed failure of one MSIV, assuming other MFIVs close (which the CTS require to be less than or equal to 60 seconds) and the peak containment pressure remains below the design limit. The containment isolation function is still available via the main feedwater isolation check valve if a MFIV is assumed to be inoperable. The feedwater isolation function is effectively present via the MFW pump trip (which is Class II but receives a Class I signal) which acts to stop feedwater flow in less than 10 seconds if the MFRV or associated bypass valve are inoperable. If the MFIV is inoperable, feedwater isolation automatically occurs via the MFRV/bypass valve closure and the MFW pump trip. The MFIV can be closed manually if required.

The CTS as written would permit a single valve in each line to be inoperable and open for the AOT, but would not permit a second valve in the flow path to be inoperable and open, and thus would require entry into 3.0.3. NUREG-1431 revises the MODES 1, 2, and 3 ACTION to include two valves in the flow path inoperable and allows up to eight hours to restore one valve to an OPERABLE condition or isolate the affected flow path. NUREG-1431 recognizes that the feedwater pump trip will effectively stop feedwater flow under these conditions. The eight hour AOT allowed by NUREG-1431 assumes that the isolation function is lost since the two valves inoperable could be in series.

As noted above, the existing plant design would still provide for containment isolation via the Class I MFW isolation check valve and provides effective feedwater isolation via the MFW pump trip even with two valves in the flow path inoperable.

IV. SPECIFIC NO SIGNIFICANT HAZARDS EVALUATION

NSHC LS16 (continued)

The proposed changes to increase the valve AOTs are consistent with recommendations from NUREG-1024, "Technical Specifications - Enhancing the Safety Impact." NUREG-1024 states in part:

Allowable outage times that are too short will subject the unit to unnecessary trips, transients and fatigue cycling. Outage times that are too short also may result in less thorough repair and post repair testing before equipment is returned to service.

Safety Function of MFIV/MFRV/Bypass Valves

The MFIV/MFRVs must be open during power operation. The valves including the bypass valves have a safety function to close to mitigate the consequences for the following:

1. The safety-related function of the MFRVs and the associated bypass valves is to provide the initial isolation of MFW flow to the secondary side of the SGs following a high energy line break (HELB) (either a feedwater (FWLB) or a MSLB line break). Since the MFRVs and associated bypass valves are located in non safety grade piping, the MFIVs ultimately provide (in less than or equal to 60 seconds) the safety-related isolation function of the MFW flow to the secondary side of the SGs following an HELB. Closure of the MFIVs or MFRVs and associated bypass valves or the automatic trip of the MFW pumps effectively terminates flow to the SGs, terminating the event for FWLBs occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by the termination of feedwater flow. The termination of feedwater to an affected SG limits the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reduces the cooldown effects for SLBs.
2. The MFIVs isolate the non safety-related portions of the piping system from the safety-related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of AFW to the intact loops. The MFW isolation check valves also function to provide a pressure boundary for the controlled addition of AFW to the intact loops.

One MFIV, one MFW isolation check valve, and one MFRV and its associated bypass valve, are located on each MFW line, outside but close to containment. The MFIVs, MFW check valves, and MFRVs are located upstream of the AFW injection point so that AFW may be supplied to the SGs following MFIV or MFRV closure.

The MFIVs and MFRVs and associated bypass valves, close on receipt of any safety injection signal (SIS), a T_{avg} low coincident with reactor trip (P-4) or SG water level high high signal. They may also be actuated manually. The MFW pumps trip on any SIS or on a SG water level high high signal. As noted above, a check valve located upstream of the MFIV is available. The check valve isolates the feedwater line, penetrating containment, and ensures that the intact SGs do not continue to feed the FWLB in the non safety-related piping upstream of the feedwater isolation check valves and that the AFW input will be forward to the SGs.

IV. SPECIFIC NO SIGNIFICANT HAZARDS EVALUATION

NSHC LS16 (continued)

3. The design basis of the MFIVs and MFRVs is established by the analyses for the large SLB and the resulting containment pressure. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs or MFRVs and associated bypass valves, are relied on to terminate an SLB for core response analysis and excess feedwater event upon the receipt of a SG water level high high signal or a feedwater isolation signal on high SG level. Failure of an MFIV, MFRV, or the associated bypass valves to close following an SLB or FWLB, can result in additional mass and energy being delivered to the SGs, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event. However, as noted above, the MFV pump trip, even though Class II, is also available to effectively terminate feedwater flow into containment.

Summary and Conclusions

The affect on unit safety of an increase in the MFIV/MFRV/Bypass valve MODE 1, 2, and 3 AOT/completion time from 4 hours to 72 hours and the addition of an eight hour AOT/completion time for two valves inoperable in a flow path to restore or close the valve(s) has been evaluated. The isolation functions are intended to protect against Condition IV events and the increase in AOT and the new ACTION related to two valves in the flowpath have minimal impact since the probabilities of these events is low. In conclusion, there is reasonable assurance the health and safety of the public will not be adversely affected by the proposed TS changes.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes to the MFIV AOT/completion times do not alter the unit configuration or operation, and consequently do not significantly increase the probability of an accident as defined in the FSAR Update. While the consequences of an accident may possibly be increased with the subject valves inoperable, the chances of an accident occurring during the AOT is very small, so the potential increase in consequences is not significant.

DCPP No Significant Hazards Evaluations

IV. SPECIFIC NO SIGNIFICANT HAZARDS EVALUATION

NSHC LS16 (continued)

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not alter any basic regulatory requirements or affect any safety analyses. Consequently, the proposed changes do not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS16" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS18
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

In MODES 5 and 6, the CTS require that the remaining CR [emergency filtration (ventilation) system (CREFS)] train be placed in the recirculation MODE if the inoperable train cannot be returned to an OPERABLE condition in seven days. NUREG-1431 recognizes the primary function of the [CREFS] in MODES 5 and 6 is to protect CR personnel from a fuel handling accident. Therefore, NUREG-1431 permits the inoperable train to remain inoperable if CORE ALTERATIONS and/or movement of irradiated fuel assemblies are suspended, thus eliminating the potential for a fuel handling accident. This is a relaxation from the CTS.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not alter the unit configuration or operation or the function of any safety system. Consequently, the change does not increase the probability of an accident as defined in the FSAR Update. Allowing the unit to comply with an ACTION REQUIREMENT by eliminating the potential for an analyzed accident places the unit in a safe condition.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS18 (continued)

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements and change any accident analysis assumptions, initial conditions, or results. Consequently, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS18" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS19
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS require that the CR [emergency filtration (ventilation) system (CREFS)] trains be tested at least once every 18 months by verifying that the system maintains the control room at a positive pressure of greater than or equal to [1/8 inches] of water gauge relative to the outside atmosphere during the pressurization MODE of operation. This testing is currently performed on both [CREFS] trains as required by the CTS. NUREG-1431 would revise the Frequency to at least once every 18 months on a STB, which would require testing only one train every 18 months. This revised testing Frequency is consistent with NUREG-0800, Section 6.4 for proving the [CR] pressure boundary integrity. The test will still evaluate the integrity of the CR structure and the ability of the [CR ventilation system] to maintain a positive pressure with respect to the outside atmosphere and adjacent areas every 18 months. The [CR ventilation system] OPERABILITY verification conducted every 31 days is not revised and will verify OPERABILITY of the system components.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC. -

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not alter the unit configuration or operation or the function of any safety system. Consequently, the change does not increase the probability of an accident as defined in the FSAR Update. Revising the testing Frequency to verify the CR pressure boundary consistent with NRC guidance does not effect the analyzed accident, its probability, or its consequences.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS19 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements or change any accident analysis assumptions, initial conditions, or results. Consequently, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS19" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS21
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The time to achieve HOT SHUTDOWN (MODE 4) if ACTIONS are not complete for the SG heat removal system is changed from 6 hours to 12 hours. The CTS have been revised per NUREG-1431 to require a SG to be OPERABLE if used for heat removal in MODE 4. The Required Action if the SG becomes inoperable due to inoperable ADV or condensate storage tank (CST), or AFW pump is to be in MODE 4 without reliance upon the SG for heat removal. Since the inoperable conditions may affect the normal cooldown capability, NUREG-1431 allows an additional six hours to reach residual heat removal (RHR) system entry conditions. The added six hours provides additional time to perform an orderly transfer from the SG method of heat removal to the RHR system to maintain the heat removal function.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not alter the unit configuration or operation or the function of any safety system. Consequently, the change does not increase the probability of an accident as defined in the FSAR Update. The proposed change permits a longer time to cooldown to RHR entry conditions; however, this is appropriate due to the reduced temperatures and pressures present and the reduced probability of an accident during these conditions.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

DCPP No Significant Hazards Evaluations

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS21 (continued)

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function. As discussed above, the change does allow additional time to complete the transfer from the SG as the method for heat removal to the RHR system, but does not alter the basic methodology.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements or change any accident analysis assumptions, initial conditions, or results.

Consequently, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS21" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS23
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A footnote is added to the SR that demonstrates that the MSIV close within five seconds to indicate that the demonstration of MSIV operability is only required to be performed for entry into (and continued operation in) MODES 1 and 2. The footnote is added in lieu of the current exception to specification 4.0.4. Exceptions to the corresponding specification in NUREG-1431 (SR 3.0.4) do not exist. While the footnote is intended to accomplish the same thing, it is in fact less restrictive because the footnote permits an indefinite stay in MODE 3 while the exception to specification 4.0.4 requires testing within 24 hours of establishing unit conditions by applying the requirements of specification 4.0.3 as discussed in GL 87-09.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes. The MSIVs are not assumed to be the initiators of any analyzed event. The deletion of the 24-hour time limit required by 4.0.4 to perform the surveillance in MODE 3 is justified on the basis of the performance of the valves during testing. Because of the high reliability of the valves and their consistent performance during testing, it is highly unlikely that one of the valves would fail to close when required even if testing were to be delayed as allowed by this revision. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation.

DCPP No Significant Hazards Evaluations

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

**NSHC LS23
(continued)**

Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change is trivial and does not alter the bases for allowing entry into MODE 3 to perform the surveillance. Any reduction in a margin of safety will be insignificant and offset by the benefit gained through avoiding an unnecessary unit transient.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS23" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS24
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change adds a new Required Action for three or more [ADV] lines inoperable and specifies a Completion Time of 24 hours to return all but two [ADV] lines to an OPERABLE status. NUREG-1431 permits two or more [ADV] lines to be inoperable with a Required Action of returning all but one to an OPERABLE status within a Completion Time of 24 hours. The 24-hour Completion Time is based upon the low probability of an event occurring that would require the [ADV] lines to function. The CTS do not specify an ACTION for three or more [ADV] lines inoperable. Therefore, the unit is required to enter (LCO) 3.0.3. The requirement to enter LCO 3.0.3 is overly restrictive based upon the NUREG-1431 allowance of 24 hours, and would subject the unit to unnecessary down-power transients.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- (3) Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not alter the unit configuration or operation or the function of any safety system. Consequently, the change does not increase the probability of an accident as defined in the FSAR Update. Revising the conditions associated with an accident mitigating system consistent with NRC guidance does not effect the analyzed accident, its probability, or its consequences. The revision simply recognizes the low probability of an event that would require the [ADV] lines to function, and that the condenser steam dump valves would likely be available.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

DCPP No Significant Hazards Evaluations

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS24 (continued)

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not alter the basic regulatory requirements or change any accident analysis assumptions, initial conditions, or results. Consequently, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS24" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS31
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS allow continued operation with inoperable MSSVs if the power range neutron flux high trip setpoints are reduced. The amount of reduction in the trip setpoint is dependent on the total number of inoperable MSSVs and is intended to compensate for the lost relieving capacity (heat removal capability) should a transient requiring their operation occur. On January 20, 1994, Westinghouse issued NSAL 94-001, which stated that the reductions in the power range neutron flux high trip setpoint required by TS Table 3.7-1 may not be bounding for the LOL/TT event since the power range neutron flux high trip setpoint specified may not be low enough to preclude secondary system overpressurization. In the proposed specification, the requirement to reduce the power range neutron flux high trip setpoint is revised such that the required setpoints satisfy the requirements of NSAL 94-001 or specific unit safety analyses. The reduction in trip setting prevents a power increase above those settings should the unit be operating with a positive MTC. The MSSVs are set to protect the secondary system against overpressurization in accordance with ASME codes and mitigate the consequences of anticipated operational transients. This change has been identified as less restrictive since the applicable set points have been revised consistent with NSAL 94-001 as opposed to NUREG-1431.

A unit may continue to operate with up to three MSSVs inoperable per steam line provided the power range neutron flux high trip setpoint is reduced as specified. Currently administrative controls are used to assure that the neutron high flux trip settings are reduced to appropriate levels per NSAL 94-001 should entry into the be required.

Upon receipt of Westinghouse NSAL 94-001, reanalysis of the LOL/TT event was performed. An analysis was performed for one MSSV inoperable on each SG that verified the plant could continue to operate at 100 percent rated thermal power. The analysis assumed worst case assumptions and that the lowest set MSSVs (the first to open during a pressure transient) were all inoperable. The analysis verified that the SG pressure would not reach 110 percent of design following a LOL/turbine trip transient. For two or three MSSVs inoperable on any SG, the algorithm recommended in Westinghouse NSAL 94-001, was used to calculate the power range neutron flux high trip setpoints. The algorithm recommended by Westinghouse is based on extremely conservative assumptions to determine the power level/steam flow that can be handled by the remaining OPERABLE MSSVs, i.e., that a reactor trip does not occur and that feedwater is unavailable. The calculation is documented and verified and specifies where the power range neutron flux high trip setpoints must be set to meet the required analysis. The required setpoints specified in Table 3.7-1 incorporate the uncertainties of the neutron flux measurement and the heat balance measurements as recommended by Westinghouse.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

DCPP No Significant Hazards Evaluations

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS31 (continued)

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety."

The following evaluation is provided for the no significant hazards consideration.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or changes to operating methodologies. This revision does not affect an accident initiator of any analyzed accident. The proposed change revises the required power level versus number of OPERABLE MSSVs to be consistent with the recommendation of, or per analyses performed in support of NSAL 94-001, which was attached to IEN 94-60.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any unit system or change the method by which any safety-related system performs its function. The increased power level and high flux trip setpoint have been evaluated for the LOL/turbine trip, which is the most significantly effected transient. The evaluation indicates that the results are within limits even with the conservative assumptions used.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed change does not change any accident analysis assumptions, initial conditions or results. Consequently, it does not have an effect on a margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS31" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS33
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would revise the ACTION for MODES 1 and 2, which currently apply to one inoperable [main feedwater isolation valve (MFIV)] to apply to one or more [MFIVs]. This is less restrictive than the current requirements which apply to only one inoperable valve.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change would have an insignificant effect on the probability of occurrence of an accident because the number of inoperable [MFIVs] would not affect any accident initiators. Therefore, the probability of an accident would not be significantly increased.

The operability of [MFIVs] could have an effect on the consequences of accidents that take credit for [MFIV] closure. However, these accidents are very low probability events that are not expected to occur during the lifetime of the unit. Nevertheless, should one of these low probability events occur, the accident analyses assume that various failures of equipment occur, and the failure of an [MFIV] is considered in these failure assumptions. Furthermore, other equipment would be expected to function to backup the [MFIV] function. These include feedwater check valves that prevent backflow through the feedwater lines, flow control valves that direct AFW flow away from a broken feedwater line, and feedwater control valves and feedwater pump trip circuits that can terminate feedwater flow. Effective decay heat removal from the unit can be accomplished with only one SG. These factors tend to mitigate the consequences of having more than one [MFIV] inoperable if an accident should occur.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS33 (continued)

Therefore, the proposed change would have no significant effect on the probability or consequences of any previously analyzed accidents.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves the number of inoperable [MFIVs]. As noted above, there is other equipment available to backup the [MFIVs] should an accident occur. This equipment assures that an accident sequence would proceed as expected and analyzed in the unit safety analysis. Although some of the backup components are not in TS, they are designed to high standards and are periodically tested to assure operability. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The change involves the operability of equipment used to mitigate postulated accidents. As noted in the evaluation of Criterion 1 above, there is backup equipment available in the design to assist in performing the [MFIV] function. This equipment is expected to operate and would perform the [MFIV] functions in sufficient time to avoid a significant reduction in any margin of safety. Therefore, the proposed change would have no significant adverse effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS33" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS37
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change exempts from the Applicability of the [MFIVs] LCO the condition where one or more [MFIV] is inoperable in MODES 1, 2, or 3, but the [MFIV, feedwater regulating (FRVs) or its associated bypass valve] is closed [and deactivated or isolated by a closed manual valve].

NUREG-1431 recognizes that if one or more [MFIVs or FRVs] or associated bypass valves are closed [and de-activated] and verified closed [], their safety function is being fulfilled and there is no need to enter the ACTION statement. If the valves cannot be closed or isolated, then NUREG-1431 requires that the unit be taken to MODE 4, which is consistent with the CTS ACTION for one [MFIV or FRV valve or associated bypass valve] that cannot be closed. Going to MODE 4 places the unit in a condition such that the LCO is no longer applicable.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or changes to operating methodologies. This revision does not affect an accident initiator of any analyzed accident. This change recognizes that since the [MFIVs or FRVs] or associated bypass valves are closed they are performing their intended safety function; consequently, there is no need to place the plant in a MODE where the LCO is not applicable.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS37 (continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not require physical alteration to any plant system or change the method by which any safety-related system performs its function. The [MFIVs, FRVs] and associated bypass valves would be in their accident mitigating position, thus fulfilling their required safety function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not change any accident analysis assumptions, initial conditions, or results. Consequently, it does not have an effect on margins of safety. Therefore, the proposed change would have no significant adverse effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS37" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS38
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change deletes the CTS ACTION "immediately suspend all operations involving ...positive reactivity changes..." associated with both CR ventilation trains being inoperable in MODES 5 and 6. The CR isolation on high radiation is designed for rapid isolation following an immediate release of radioactivity. CORE ALTERATIONS, positive reactivity additions via deboration and movement of irradiated fuel are processes that could lead to the release of radioactivity. However, CORE ALTERATIONS and movement of irradiated fuel are the only processes that could lead to an immediate release of radioactivity. Positive reactivity addition via deboration is a slow process and does not represent a potential for an immediate release of radioactivity. The deletion of this ACTION is consistent with NUREG-1431.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not result in any hardware changes or changes to operating methodologies. This change recognizes that since a positive reactivity addition via deboration is a slow change that the immediate suspension of such a positive reactivity change is beyond the basis of the LCO.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS38 (continued)

The proposed change does not require physical alteration to any plant system or change the method by which any safety-related system performs its function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not change any accident analysis assumptions, initial conditions, or results. Consequently, it does not have an effect on margins of safety. Therefore, the proposed change would have no significant adverse effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS38" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

V. RECURRING NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC TR1 10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the CTS to allow the use of actual actuation signals for SRs that currently require testing using simulated test signals only. This change is consistent with NUREG-1431.

In several SRs in the CTS, operability of certain equipment is demonstrated by ensuring that the equipment performs its safety function upon receipt of a simulated test signal. The intent of a "simulated" signal was to be able to perform the required testing without the occurrence (or without causing) an actual signal generating event. However, the unintended effect was to require the performance of the SR (using a test signal) even if an actual signal had previously verified the operation of the equipment. This change allows credit to be taken for actual events when the required equipment actuates successfully.

While the occurrence of events that cause actuation of accident mitigation equipment is undesirable, the actuation of mitigation equipment on an actual signal is a better demonstration of its operability than an actuation using a test signal. Thus, the change does not reduce the reliability of the equipment tested. The change also improves unit safety by reducing the amount of time the equipment is taken out of service for testing, and thereby increasing its availability during an actual event, and by reducing the wear of the equipment caused by unnecessary testing.

This proposed TS change has been evaluated and it has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change allows the use of an actual actuation signal (when/if it occurs) to satisfy SRs currently requiring simulated test signals to demonstrate equipment operability. While the change credits events that may have occurred, it has no adverse effect on any accident initiators

V. RECURRING NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC TR1
(continued)

or accident consequences. In fact, by potentially reducing unnecessary testing, it may reduce the probability of an accident because the testing itself can increase the probability of an accident. It may also reduce accident consequences by increasing the equipment availability (i.e., less time in test).

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not involve a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The use of an actual actuation signal to satisfy a SR either has no impact on, or increases the margin of unit safety by:

- a) Increasing mitigation equipment availability, and,
- b) Improving mitigation equipment reliability by potentially reducing wear caused by unnecessary testing.

The change is consistent with the safety analysis and licensing basis.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Therefore, it is concluded that, based on the above safety evaluation, the activities associated with NSHC "TR1" satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

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INDUSTRY TRAVELERS APPLICABLE TO SECTION 3.7

TRAVELER NUMBER	STATUS	JUSTIFICATION NUMBER	COMMENTS
TSTF-36 Rev.2	Incorporated.	3.7-42; only applicable to DCPD.	Adds Note that states that LCO 3.0.3 is not applicable.
TSTF-44	Not Incorporated.	Not applicable.	Retained CTS requirements.
TSTF-51	Not Incorporated.	Not applicable.	Requires plant specific re-analysis to establish decay time dependence for fuel handling accident.
TSTF-70 Rev. 1	Not Incorporated.	Not applicable.	Not NRC approved as of traveler cut-off date
TSTF-100	Incorporated.	3.7-05 and 3.7-19	NRC approved.
TSTF-101	Incorporated.	3.7-29	NRC approved.
TSTF-102	Not Incorporated.	Not applicable	NRC rejected change.
WOG-83 (Formerly WOG-31)	Incorporated.	3.7-01	Revises MSSV TS to require flux trip setpoint reduction for plants licensed with a positive MTC.
WOG 64	Incorporated by TU, UE & WC. Not Incorporated by DCPD as current licensing basis.	3.7-34 for TU, UE & WC. Not applicable to DCPD.	Revises MSIV TS to allow 72 hours for AOT for one inoperable MSIV.
TSTF-139 (CEOG-51)	Incorporated.	Not Applicable- Bases change.	ITS LCO 3.7.15 Bases change only.
TSTF-140 (CEOG-52)	Not Incorporated.	Not Applicable.	Editorial changes to ITS LCO 3.7.6 deemed unnecessary for LCO understanding (level vs volume).
WOG-86	Incorporated by TU, UE & WC. Not Incorporated by DCPD as current licensing basis.	3.7-57 for TU, UE & WC. Not applicable to DCPD.	Provides new ACTION for failure of a ventilation system envelop SR.
WOG-98	Incorporated.	3.7-56.	Revises MSIV [and MFIV, MFRV and bypass valve] testing to separate IST and automatic actuation requirements .

NUREG-1431 SPECIFICATIONS WHICH ARE NOT APPLICABLE

<u>Specification #</u>	<u>Specification Title</u>	<u>Comments</u>
3.7.11	Control Room Emergency Air Temperature Control System (CREATCS)	
3.7.14	Penetration Room Exhaust Air Cleanup System (PREACS)	

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LC0 3.7.1 The Five MSSVs per steam generator shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

3.7-01

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more required MSSVs inoperable.</p>	<p>A.1 Reduce power Thermal Power to less than or equal to the applicable Maximum Allowable % RTP listed specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.</p> <p><u>AND</u></p> <p>A.2 ----- NOTE ----- Only applicable in MODE 1</p> <p>-----</p> <p>Reduce the Power Range High Neutron Flux trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.</p>	<p>4 hours</p> <p><u>3.7-01</u></p> <p>72 hours</p> <p><u>3.7-01</u></p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Required Action and associated Completion Time not met.</p> <p><u>OR</u></p> <p>One or more steam generators with less than two MSSVs OPERABLE.</p>	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Be in MODE 4.</p>	<p>6 hours</p> <p>12 hours</p> <p><u>B.</u></p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.1.1</p> <p>-----NOTE----- Only required to be performed in MODES 1 and 2. -----</p> <p>Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within $\pm 1\%$.</p>	<p>In accordance with the Inservice Testing Program</p>

Table 3.7.1-1 (page 1 of 1)
 OPERABLE Main Steam Safety Valves versus
 Applicable ~~Maximum Allowable~~ Power in Percent of RATED THERMAL POWER

MINIMUM NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM ALLOWABLE (RTP)
5	100
4	65 87
3	46 47
2	29 29

~~B=PS~~

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 ~~Four~~ MSIVs shall be OPERABLE.

B

APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed and ~~de-activated~~.

B

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	8 hours <u>B</u>
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
C. -----NOTE----- Separate Condition entry is allowed for each MSIV. ----- One or more MSIVs inoperable in MODE 2 or 3.	C.1 Close MSIV. <u>AND</u> C.2 Verify MSIV is closed.	8 hours <u>B</u> Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.2.1 -----NOTE----- Only required to be performed in MODES 1 and 2. ----- Verify closure time of each MSIV is \leq [4.6] seconds on an actual or simulated actuation signal.</p>	<p>B-PS B-PS 3.7-56 In accordance with the [Inservice Testing Program or 18 months]</p>
<p>SR 3.7.2.2 -----NOTE----- Only required to be performed in MODES 1 and 2. ----- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.</p>	<p>3.7-56 18 months</p>

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves B

LCO 3.7.3 Four MFIVs, four MFRVs, and associated bypass valves shall be OPERABLE. B

APPLICABILITY: MODES 1, 2, and 3 except when MFIV, MFRV, or associated bypass valve is closed and deactivated or isolated by a closed manual valve. B

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFIVs inoperable.	A.1 Close or isolate MFIV. <u>AND</u> A.2 Verify MFIV is closed or isolated.	72 hours <u>B</u> Once per 7 days
B. One or more MFRVs inoperable.	B.1 Close or isolate MFRV. <u>AND</u> B.2 Verify MFRV is closed or isolated.	72 hours <u>B</u> Once per 7 days
C. One or more MFRV or preheater bypass valves inoperable.	C.1 Close or isolate bypass valve. <u>AND</u> C.2 Verify bypass valve is closed or isolated.	72 hours B-PS <u>B</u> Once per 7 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two valves in the same flow path inoperable.	D.1 Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours <u>B</u>
	E.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.3.1 Verify the closure time of each MFIV is is 60 seconds. MFRV [and associated bypass valve] is is [7] seconds on an actual or simulated actuation signal.</p>	<p><u>3.7-3</u></p> <p>In accordance with the [Inservice Testing Program or [18] months] B-PS <u>3.7-56</u></p>
<p>SR 3.7.3.2 Verify the closure time of each MFRV and associated bypass valve is is 7 seconds.</p>	<p>At each COLD SHUTDOWN but not more frequently than once per 92 days <u>3.7-3</u></p>
<p>SR 3.7.3.3 Verify each MFIV actuates to the isolation position on an actual or simulated actuation signal.</p>	<p><u>18 months</u> <u>3.7-56</u></p>

3.7 PLANT SYSTEMS

3.7.4 ~~10%~~ Atmospheric Dump Valves (ADVs)

PS

LCO 3.7.4 ~~Three~~ ~~Four~~ ADV lines shall be OPERABLE.

B-PS

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ADV line inoperable.	A.1 -----NOTE----- LCO 3.0.4 is not applicable. ----- Restore required ADV line to OPERABLE status.	7 days
B. Two or more required ADV lines inoperable.	B.1 Restore all but one ADV line to OPERABLE status.	24 72 hours <u>3.7-5</u> <u>3.7-6</u>
C. Three or more required ADV lines inoperable	C.1 Restore all but two ADV lines to OPERABLE status	24 hours <u>3.7-5</u> <u>3.7-6</u>
C D. Required Action and associated Completion Time not met.	C D. 1 Be in MODE 3. <u>AND</u> C D. 2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours <u>3.7-6</u> 18 hours <u>B</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.4.1 Verify one complete cycle of each ADV.	18 months <u> B </u>
SR 3.7.4.2 Verify one complete cycle of each ADV block valve.	18 months <u> B </u>
SR 3.7.4.3 Verify that the backup air bottle for each ADV has a pressure ≥ 260 psig.	24 hours <u> 3.7-50 </u>

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

B

LCO 3.7.5 ~~Three~~ AFW trains shall be OPERABLE.

~~NOTE~~
~~Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.~~

B

APPLICABILITY: MODES 1, 2, and 3,
 MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days <u>B</u> AND 10 days from discovery of failure to meet the LCO
B. One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours <u>B</u> AND <u>B</u> 10 days from discovery of failure to meet the LCO

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Required Action and associated Completion Time for Condition A or B not met.</p> <p><u>OR</u></p> <p>Two AFW trains inoperable in MODE 1, 2, or 3.</p>	<p>C.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>C.2 Be in MODE 4.</p>	<p>6 hours</p> <p><u>B</u></p> <p><u>B</u></p> <p>18 hours</p>
<p>D. Three AFW trains inoperable in MODE 1, 2, or 3.</p>	<p>D.1 <u>NOTE</u></p> <p>LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status.</p> <p>Initiate action to restore one AFW train to OPERABLE status.</p>	<p><u>B</u></p> <p><u>B</u></p> <p>Immediately</p>
<p>E. Required AFW train inoperable in MODE 4.</p>	<p>E.1 Initiate action to restore AFW train to OPERABLE status.</p>	<p>Immediately</p>
<p>F. With the GST flow path not open to the AFW pump suction.</p>	<p>F.1 Restore the GST flow path.</p>	<p>4 hours</p> <p><u>3.7-9</u></p>
<p>G. With the FWST flow path not capable of being aligned to the AFW pump suction.</p>	<p>G.1 Restore the capability of FWST alignment.</p>	<p>7 days</p> <p><u>3.7-9</u></p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
Required Action and associated Completion Time for Condition F or G not met	H.1 Be in MODE 3	6 hours
	AND H.2 Be in MODE 4 without reliance upon steam generator for heat removal.	3-7-9 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.5.1 Verify each AFW manual, power operated, and automatic valve in each water flow path, [and in both steam supply flow paths to the steam turbine driven pump,] that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days <u>B</u>
SR 3.7.5.2 NOTE Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ [1000] 650 psig in the steam generator Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Test Program 3-7-29 <u>B</u> [31] days on a STAGGERED TEST BASIS <u>B-PS</u>
SR 3.7.5.3 NOTE Not applicable in MODE 4 when steam generator is relied upon for heat removal. Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	<u>B</u> 18 months

Surveillance Requirements (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.7.5.4</p> <p>-----NOTES-----</p> <p>1. Not required to be performed for the turbine driven AFW pump until 24 hours after \geq [1000] 650 psig in the steam generator.</p> <p>2. Not applicable in MODE 4 when generator is relied upon for heat removal.</p> <p>-----</p> <p>Verify each AFW pump starts automatically on an actual or simulated actuation signal.</p>	<p><u>B</u></p> <p><u>B-PS</u></p> <p>18 months <u>B</u></p>
<p>SR 3.7.5.5 Not Used</p> <p>Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.</p>	<p><u>B-PS</u></p> <p>Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for > 30 days</p>
<p>SR 3.7.5.6 Verify the FWST is capable of being aligned to the AFW system by cycling each FWST valve in the flow path necessary for realignment through at least one full cycle.</p>	<p>92 days</p> <p><u>3.7-9</u></p>

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST) and Fire Water Storage Tank (FWST)

3.7-10

LC0 3.7.6 The CST level shall be \geq ~~[110,000 gal]~~ 41.3% and the FWST level shall be \geq 22.2% for one unit operation and \geq 41.7% for two unit operation.

3.7-10

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST level not within limit.	A.1 Verify by administrative means OPERABILITY of backup water supply. Restore CST level to within limit.	4 hours <u>AND</u> <u>3.7-10</u> Once per 12 hours thereafter
B. FWST level not within limit.	<u>AND</u> A.2B.1 Restore CST FWST level to within limit.	<u>3.7-10</u> 7 d ays
B.C. Required Action and associated Completion Time not met.	B.C.1 Be in MODE 3. <u>AND</u> B.C.2 Be in MODE 4, without reliance on steam generator for heat removal.	6 hours 18 hours <u>B.C.</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CST level is \geq [110,000 gal] 41.3 %.	12 hours <u>3.7-10</u>
SR 3.7.6.2	Verify the FWST level is \geq 22.2 % for one unit operation and \geq 41.7 % for two unit operation.	12 hours <u>3.7-10</u>

3.7 PLANT SYSTEMS

3.7.7 ~~Vital~~ Component Cooling Water (CCW) System

PS

LCO 3.7.7 Two ~~vital~~ CCW trains ~~loops~~ shall be OPERABLE.

PS

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One vital CCW train loop inoperable.	A.1 -----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW. ----- Restore vital CCW train loop to OPERABLE status.	<u>PS</u> 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.7.1 -----NOTE----- Isolation of CCW flow to individual components does not render the CCW System inoperable -----</p> <p>Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.</p>	<p>31 days</p>
<p>SR 3.7.7.2 Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.</p>	<p>18 months <u>B</u></p>
<p>SR 3.7.7.3 Verify each CCW pump starts automatically on an actual or simulated actuation signal.</p>	<p>18 months <u>B</u></p>

3.7 PLANT SYSTEMS

3.7.8 Service Water ~~Auxiliary Saltwater (ASW)~~ System (SWS)

PS

LCO 3.7.8 Two SWS ~~ASW~~ trains shall be OPERABLE.

PS

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One SWS ASW train inoperable.</p>	<p>A.1 -----NOTES----- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AG Sources - Operating," for emergency diesel generator made inoperable by SWS.</p> <p>Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by SWS ASW.</p> <p>-----</p> <p>Restore SWS ASW train to OPERABLE status.</p>	<p><u>3.7-13</u></p> <p><u>PS</u></p> <p>72 hours</p>

Surveillance Requirements (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.8.1</p> <p style="text-align: center;"><u>NOTE</u></p> <p>Isolation of SWS flow to individual components does not render the SWS inoperable.</p> <p>Verify each SWS ASW manual- and power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position or that a motive force is available such that the valve would be capable of being placed in the correct position.</p>	<p>3.7-14</p> <p>PS</p> <p>31 days</p> <p>3.7-15</p>
<p>SR 3.7.8.2</p> <p>Verify each SWS automatic ASW power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to can be moved to the correct position. on an actual or simulated actuation signal</p>	<p>18 months</p> <p>B-PS</p> <p>PS</p> <p>3.7-16</p> <p>In accordance with the Inservice Test Program.</p>
<p>SR 3.7.8.3</p> <p>Verify each SWS ASW pump starts automatically on an actual or simulated actuation signal.</p>	<p>18 months</p> <p>B</p> <p>PS</p>

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more cooling towers with one cooling tower fan with the UHS inoperable	A.1 Restore cooling tower fan(s) to OPERABLE status. Place a second CCW heat exchanger in service	7 days 8 hours <u>B</u> <u>3.7-17</u>
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> UHS inoperable [for reasons other than Condition A].	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours <u>B</u> 36 hours <u>3.7-17</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.9.1 NOT USED Verify water temperature level of UHS is $\leq 60^{\circ}\text{F}$ [] $> [562]$ ft [mean sea level]	[24] hours <u>3.7-17</u>
SR 3.7.9.2 If water temperature is $> 60^{\circ}\text{F}$ but $\leq 62^{\circ}\text{F}$ [] Verify average water temperature of UHS is $\leq [90]^{\circ}\text{F}$ within limits.	24 hours <u>B</u> if UHS temperature is $\leq 60^{\circ}\text{F}$ AND <u>B-PS</u> 12 hours if UHS temperature $> 60^{\circ}\text{F}$ but $\leq 62^{\circ}\text{F}$ AND 2 hours if UHS temperature $> 62^{\circ}\text{F}$ but $\leq 64^{\circ}\text{F}$ <u>3.7-17</u>
SR 3.7.9.3 Operate each cooling tower fan for $\geq [15]$ minutes.	31 days <u>B-PS</u>
SR 3.7.9.4 Verify each cooling tower fan starts automatically on an actual or simulated actuation signal.	18 months <u>B-PS</u>

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Filtration ~~Ventilation~~ System (GREFS
CRVS)

PS

LCO 3.7.10 Two GREFS ~~CRVS~~ trains shall be OPERABLE.

PS

APPLICABILITY: MODES 1, 2, 3, 4, ~~5~~ and ~~6~~
During movement of irradiated fuel assemblies,
~~[During CORE ALTERATIONS].~~

B

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One GREFS CRVS train inoperable.	A.1 Restore GREFS CRVS train to OPERABLE status.	7 days <u>PS</u>
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6 or during movement of irradiated fuel assemblies [or during CORE ALTERATIONS].	C.1 <u>NOTE</u> Place in toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable. Place OPERABLE GREFS CRVS train in emergency recirculation mode. <u>OR</u>	<u>B-PS</u> <u>B</u> Immediately <u>PS</u>

ACTIONS (continued)

CRVS
3.7.10

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1 Suspend CORE ALTERATIONS	Immediately <u>B</u>
	AND C.2.2 Suspend movement of irradiated fuel assemblies.	Immediately <u>B</u>
D. Two GREFS CRVS trains inoperable for reasons other than Condition D in MODE 5 or 6, or during movement of irradiated fuel assemblies [or during CORE ALTERATIONS].	D.1 Suspend CORE ALTERATIONS	Immediately <u>B</u>
	AND D.2 Suspend movement of irradiated fuel assemblies.	Immediately <u>B-PS</u> <u>PS</u>
E. Two GREFS CRVS trains inoperable for reasons other than Condition D in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately <u>PS</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.10.1 Operate each GREFS CRVS train for ≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days <u>B-PS</u> <u>PS</u>
SR 3.7.10.2 Perform required GREFS CRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP <u>PS</u> <u>B</u>

3.7 PLANT SYSTEMS

3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

3.7-52

LCO 3.7.11 ~~Two CREATCS trains shall be OPERABLE. Not Used~~

APPLICABILITY: ~~MODES 1, 2, 3, 4, [5, and 6,]
During movement of irradiated fuel assemblies,
[During CORE ALTERATIONS].~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREATCS train inoperable.	A.1 Restore CREATCS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met [in MODE 5 or 6, or] during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS].	C.1 Place OPERABLE CREATCS train in operation. OR [C.2.1] Suspend CORE ALTERATIONS. AND C.2.[2] Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately
D. Two CREATCS trains inoperable [in MODE 5 or 6, or] during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS].	D.1 Suspend CORE ALTERATIONS. AND D.[2] Suspend movement of irradiated fuel assemblies.	Immediately Immediately

ACTIONS (continued) CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two CREATCS trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.11.1 Verify each CREATCS train has the capability to remove the assumed heat load.	[18] months

3.7 PLANT SYSTEMS

3.7.12 ~~Emergency Core Cooling System (ECCS) Pump Room Exhaust
Air Cleanup System (PREACS) Auxiliary Building
Ventilation System (ABVS)~~ PS PS
PS

LCO 3.7.12 Two ~~ECCS PREACS~~ ABVS trains shall be OPERABLE.
PS

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The common HEPA filter and/or charcoal adsorber inoperable.	A.1 Restore the common HEPA filter and charcoal adsorber to OPERABLE status.	24 hours 3.7-21
A B. One ECCS PREACS <u>ABVS</u> train inoperable.	A B.1 Restore ECCS PREACS <u>ABVS</u> train to OPERABLE status.	7 days <u>PS</u>
C. ECCS Pump Room Pressure not capable of being maintained \leq [0.125] inches water gauge in the [post accident] mode of operation.	C.1 Restore capability to maintain pressure in the penetration room within limit.	24 hours 3.7-23
B C. Required Action and associated Completion Time not met.	B C.1 Be in MODE 3. <u>AND</u> B C.2 Be in MODE 5.	6 hours 36 hours <u>B</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.12.1	Operate each EGCS PREACS ABVS train for ≥ 15 minutes, and one train for ≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days <u>PS</u> <u>3.7-21</u>
SR 3.7.12.2	Perform required EGCS PREACS ABVS filter testing in accordance with the Ventilation Filter Testing Program (VFETP).	In accordance with the VFETP <u>PS</u> <u>B</u>
SR 3.7.12.3	Verify each EGCS PREACS ABVS train actuates on an actual or simulated actuation signal and the system realigns to exhaust through the common HEPA filter and charcoal adsorber.	18 months <u>PS</u> <u>B</u> <u>3.7-22</u>
SR 3.7.12.4	NOT USED Verify one EGCS PREACS train can maintain a pressure $\leq [0.125]$ inches water gauge relative to atmospheric pressure during the [post accident] mode of operation at a flow rate of $\leq [3000]$ cfm.	[18] months on a STAGGERED TEST BASIS <u>3.7-23</u>
SR 3.7.12.5	NOT USED Verify each EGCS PREACS filter bypass damper can be closed.	[18] months <u>3.7-24</u>
SR 3.7.12.6	Verifying that leakage through the ABVS Dampers M2A and M2B is less than or equal to 5 cfm when subjected to a Constant Pressure or Pressure Decay Leak Rate Test in accordance with ASME N510-1989. The test pressure for the leak rate test shall be based on a maximum operating pressure as defined in ASME N510-1989, of 8 inches water gauge.	18 months <u>3.7-18</u> <u>PS</u>

3.7 PLANT SYSTEMS

3.7.13 Fuel Handling Building Air Cleanup Ventilation System (FBACS FHBVS) PS

LC0 3.7.13 Two FBACS FHBVS trains shall be OPERABLE. PS

APPLICABILITY: ~~[MODES 1, 2, 3, and 4.]~~
During movement of irradiated fuel assemblies in the fuel handling building. B-PS

NOTE
LC0 3.0.3 is not applicable

PS
3.7-42

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One FBACS FHBVS train inoperable.	A.1 Restore FBACS FHBVS train to OPERABLE status. <u>OR</u>	7 days <u>Immediately</u> <u>PS</u>
	A.2 Place the OPERABLE FHBV train in operation. <u>OR</u>	Immediately <u>3.7-43</u>
	A.3 Suspend movement of irradiated fuel assemblies in the fuel handling building.	<u>Immediately</u>
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4. <u>OR</u> Two FBACS trains inoperable in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours <u>3.7-43</u> 36 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time [of Condition A] not met during movement of irradiated fuel assemblies in the fuel building.	C.1 Place OPERABLE FBACS train in operation. OR C.2 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately Imme <u>3.7-43</u> diat ely
DD. Two FBACS FHBVS trains inoperable during movement of irradiated fuel assemblies in the fuel building.	DD.1 Suspend movement of irradiated fuel assemblies in the fuel handling building.	Immediately <u>PS</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.13.1 Operate each FBACS FHBVS train for [> 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].	31 day <u>PS</u> s <u>B-PS</u>
SR 3.7.13.2 Perform required FBACS FHBVS filter testing in accordance with the <u>Ventilation Filter Testing Program (VFTP)</u> .	In accordance with the <u>VFTP</u> <u>B</u> <u>PS</u>
SR 3.7.13.3 Verify each FBACS FHBVS train actuates on an actual or simulated actuation signal.	18 months <u>B</u> <u>PS</u>
SR 3.7.13.4 Verify one FBACS FHBVS train can maintain a pressure ≤ 0.125 inches water gauge with respect to atmospheric pressure during the post accident mode of operation at a flow rate ≤ [20,000] cfm.	18 mon <u>3.7-49</u> ths <u>B</u> on a <u>PS</u> STAGGERED TEST BASIS

SURVEILLANCE	FREQUENCY
SR 3.7.13.5 Not Used Verify each FBACS filter bypass damper can be closed.	[18] months

3.7 PLANT SYSTEMS

3.7-55

3.7.14 Penetration Room Exhaust Air Cleanup System (PREACS)

LCO 3.7.14 ~~Not Used~~ Two PREACS trains shall be OPERABLE.

APPLICABILITY: ~~MODES 1, 2, 3, and 4.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PREACS train inoperable.	A.1 Restore PREACS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Operate each PREACS train for [> 10 continuous hours with heaters operating or (for systems without heaters) > 15 minutes].	31 days
SR 3.7.14.2 Perform required PREACS filter testing in accordance with the [Ventilation Filter Testing Program (VFTP)].	In accordance with the [VFTP]

(continued)

SURVEILLANCE	FREQUENCY
SR 3.7.14.3 Verify each PREACS train actuates on an actual or simulated actuation signal.	[18] months
SR 3.7.14.4 Verify one PREACS train can maintain a pressure < [0.125] inches water gauge relative to atmospheric pressure during the [post accident] mode of operation at a flow rate of < [3000] cfm.	[18] months on a STAGGERED TEST BASIS
SR 3.7.14.5 Verify each PREACS filter bypass damper can be closed.	[18] months

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Storage Pool Water Level

PS

LCO 3.7.15 The spent fuel storage pool water level shall be \geq 23 ft over the top of irradiated fuel assemblies seated in the storage racks. PS

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool. PS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1 -----NOTE----- LCO 3.0.3 is not applicable. ----- Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.	<div style="text-align: right;"><u>PS</u></div> Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.15.1 Verify the spent fuel storage pool water level is \geq 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days <u>PS</u>

~~Spent~~ Fuel Storage Pool Boron Concentration
3.7.16

3.7 PLANT SYSTEMS

3.7.16 ~~Spent~~ Fuel Storage Pool Boron Concentration

PS

LCO 3.7.16 The ~~spent~~ fuel storage pool boron concentration shall be B-PS
~~≥ [2300]~~
~~2000~~ ppm.

APPLICABILITY: When fuel assemblies are stored in the ~~spent~~ fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage.

PS
3.7-53
PS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool boron concentration not within limit.	<p style="text-align: center;">-----NOTE----- LCO 3.0.3 is not applicable. -----</p> <p>A.1 Suspend movement of fuel assemblies in the spent fuel storage pool.</p> <p style="text-align: center;"><u>AND</u></p> <p>A.2.1 Initiate action to restore spent fuel storage pool boron concentration to within limit.</p> <p style="text-align: center;"><u>OR</u></p> <p>A.2.2 Verify by administrative means [Region 2] fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool.</p>	<p style="text-align: right;"><u>PS</u></p> <p style="text-align: right;">Immediately</p> <p style="text-align: right;">Immediately</p> <p style="text-align: right;">Immediately <u>3.7-53.</u></p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.16.1 Verify the spent fuel storage pool boron concentration is within limit.	<p style="text-align: right;">7 31 da ys <u>PS</u> <u>3.7-53</u></p>

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

3.7-51

ECO 3.7.17.1 The combination of initial enrichment, initial B-10 content, burnup, and storage pattern of each spent fuel assembly stored in Region 1 shall be:

- a) The initial enrichment is 4.5 weight percent U-235 or less, or
- b) The initial enrichment is from 4.5 up to a maximum of 5.0 weight percent U-235 and any of the following conditions are met:
 - 1) The combination of initial enrichment and cumulative burnup of the assemblies is within the acceptable area of Figure 3.7.17-1, or
 - 2) The assemblies initially contained a minimum of a nominal 36 mg/in. per assembly of the isotope B-10 integrated in the fuel rods, or
 - 3) The assemblies are put in a checkerboard pattern with any of the following:
 - a) water cells, or
 - b) assemblies that initially contained a minimum of a nominal 72 mg/in. per assembly of the isotope B-10 integrated in the fuel rods, or
 - c) partially irradiated fuel of at least 8000 MWD/MTU cumulative burnup, or
 - 4) The assemblies are put into a pattern with alternate rows of fuel assemblies and water cells.

APPLICABILITY: Whenever any fuel assembly is stored in Region 1 of the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	<p>A.1</p> <p>----- NOTE ----- LCO 3.0.3 is not applicable.</p> <p>Initiate action to move the noncomplying fuel assembly into an acceptable pattern that complies with this LCO or LCO 3.7.17.2.</p>	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.17.1.1 Verify by administrative means that the fuel assembly characteristics and its expected location is in accordance with LCO 3.7.17.1.	Prior to each fuel assembly move, when the assembly will be stored in Region 1.

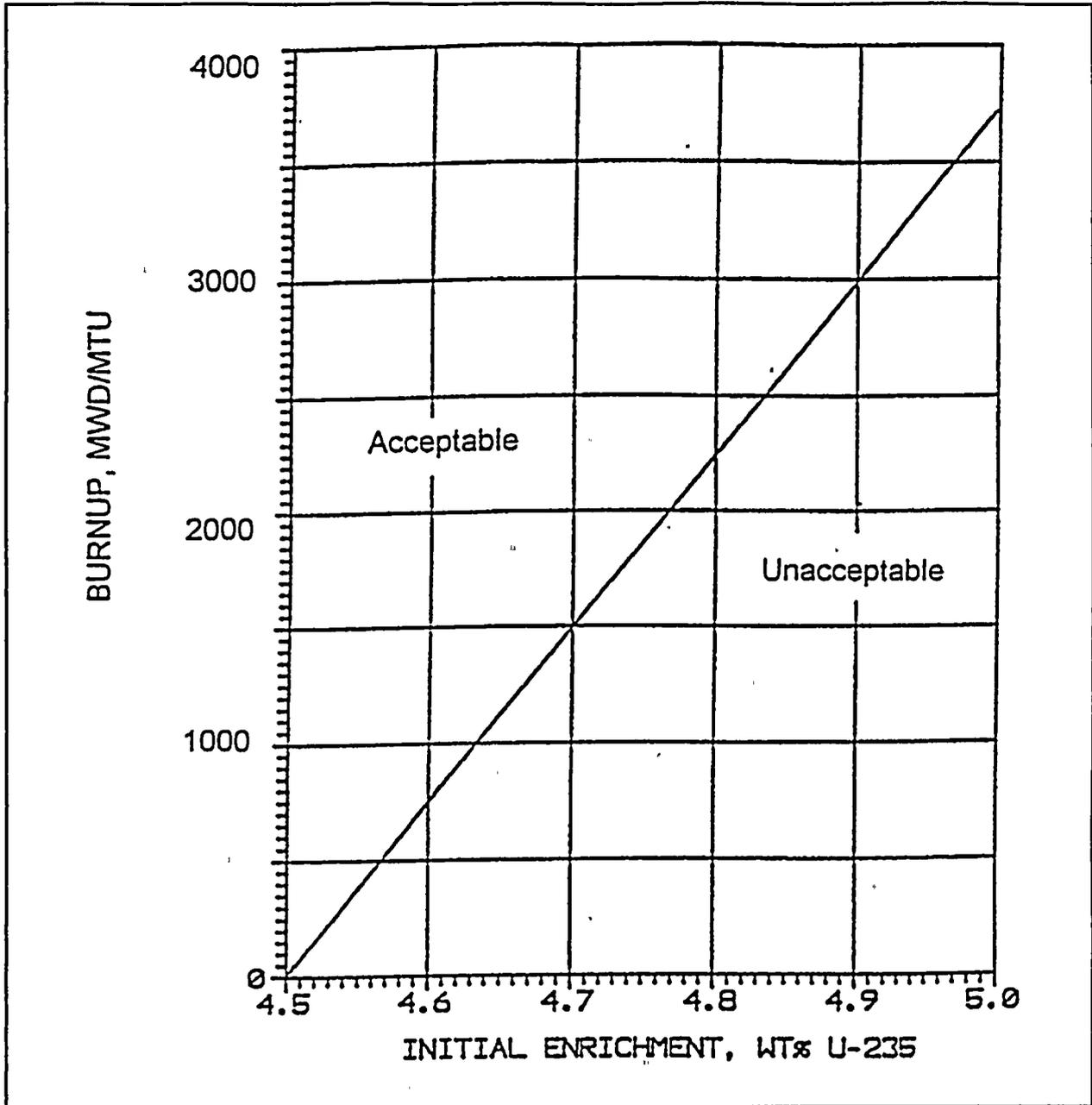


FIGURE 3.7.17-1
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT (NO IFBA) TO PERMIT
STORAGE IN REGION 1

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17.2 The combination of initial enrichment, fuel pellet diameter and burnup of each spent fuel assembly stored in Region 2 shall be within the acceptable area of Figure 3.7.17-2 or the fuel assembly is stored in a checkerboard pattern with water cells or non-fissile material in accordance with Specification 4.3.1.1. 3.7-54
B

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 of the spent fuel storage pool. B
PS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 -----NOTE----- LCO 3.0.3 is not applicable. ----- Initiate action to move the noncomplying fuel assembly from [Region 2] into an acceptable pattern that complies with this LCO or LCO 3.7.17.1.	<u>3.7-54</u> Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.17.2.1 Verify by administrative means that the initial enrichment and burnup of the fuel assembly characteristics and its expected storage location are in accordance with Figure 3.7.17-2 or Specification 4.3.1.1 LCO 3.7.17.2.	<u>3.7-54</u> Prior to storing each fuel assembly move when the assembly will be stored in Region 2. <u>B</u>

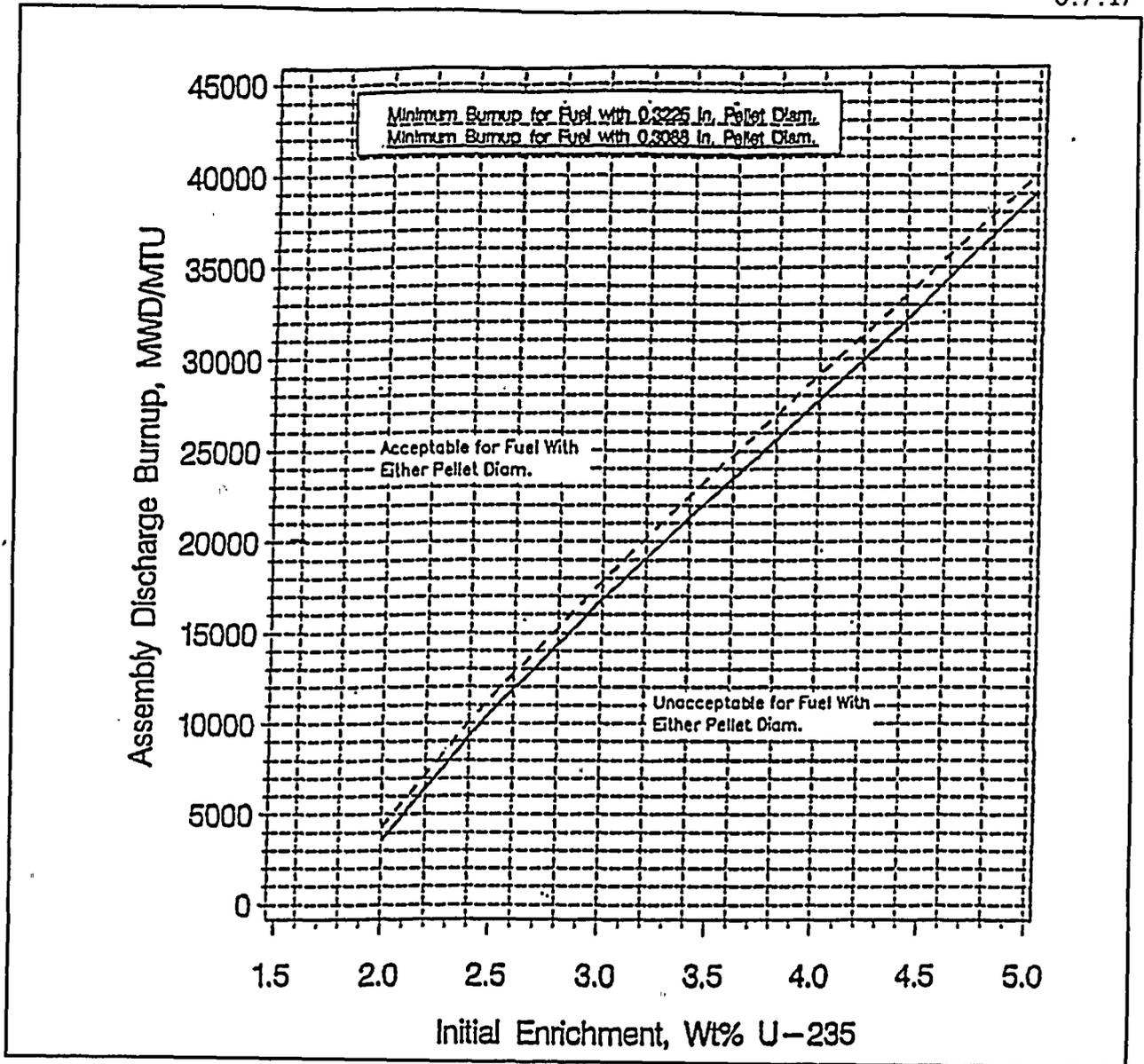


FIGURE 3.7.17.2
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT
STORAGE IN REGION 2

3.7 PLANT SYSTEMS

3.7.18 Secondary Specific Activity

LCO 3.7.18 The specific activity of the secondary coolant shall be B
 $\leq 0.10 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours
	<u>AND</u> A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.18.1 Verify the specific activity of the secondary coolant is $\leq 0.10 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$.	31 days <u>B</u>

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes- Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

**Methodology For Mark-up of NUREG-1431 Specifications
(Continued)**

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES CONTENTS

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B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section [10.3.1] (Ref. 1). The MSSV capacity criteria is 110% of rated steam flow at must have sufficient capacity to limit the secondary system pressure to $\leq 110\%$ of the steam generator design pressure. This meets the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip during an overpressure event.

APPLICABLE
SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to $\leq 110\%$ of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section [15.2] and [15.3] (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO with respect to secondary system pressure. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. If a minimum reactivity feedback is assumed, the reactor is tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer relief valves and sprays. The analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

~~If maximum reactivity feedback is assumed, the reactor is tripped on overtemperature ΔT . The departure from nucleate boiling ratio increases throughout the transient, and never drops below its initial value. Pressurizer relief valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems. The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.~~

The MSSVs satisfy Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

The accident analysis requires ~~four~~ ~~that~~ ~~five~~ MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. An MSSV will be considered inoperable if it fails to open on demand. The LCO requires that ~~five~~ MSSVs per steam generator be OPERABLE in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1.1 in the accompanying LCO, and Required Action A.2.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseal when pressure has been reduced. The OPERABILITY of the MSSVs is determined/verified by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

APPLICABILITY In MODE 1 above [40] % RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. Below [40%] in MODES 1, 2, and 3, only two [five] MSSVs per steam generator are required to be OPERABLE to limit secondary pressure.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 [and A.2]

With one or more MSSVs inoperable, reduce power action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Continued operation with less than all [five] MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER [and the Power Range Neutron Flux trip setpoint] so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. For example, if one MSSV is inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 20%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced by at least 20%. This is accomplished by reducing THERMAL POWER by at least 20%, which conservatively limits the energy transfer to all steam generators to approximately 80% of total capacity, consistent with the relief capacity of the most limiting steam generator. If one MSSV is inoperable on a SG, calculation N-114 (Ref 8) demonstrates via RETRAN analysis that the secondary system pressure peak resulting from the limiting AOO is < 110% of design. The transient is terminated by a reactor trip, either high pressurizer pressure or OTDT, and the MSSVs maintain steam pressure below 110% at design

When a MSSV(s) is inoperable, the power must be reduced in 4 hours (required action A.1) to a value less than or equal to the value specified in table 3.7.1-1, corresponding to the number of OPERABLE MSSVs regardless of the value of the MTC.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

The Power Range Neutron Flux-high trip setpoint must also be reduced in 72 hours (Required by Action A-2), to less than or equal to the value specified in Table 3.7.1-1, corresponding to the number of OPERABLE MSSVs regardless of the value of the MIT. Required Action A-2 is modified by a Note. The Note indicates that the Power Range Neutron Flux-high trip setpoint reduction is only required in MODE 1. In MODE 1, a reduced Power Range Neutron Flux-high trip setpoint provides the required protection. In MODES 2 and 3, the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation" provide sufficient protection. Thus, reduction of the Power Range Neutron Flux-high trip setpoint is not necessary in MODE 2 or 3.

The algorithm and the RETRAN analysis used for References 7 and 9 are conservative since they both assume that the relief capacity is accordingly reduced on each SG and that the flow capacity for all inoperable MSSVs is that of the highest capacity valve.

The calculated power level is further reduced to account for instrument and channel and heat balance uncertainties and is the value specified as the MAXIMUM ALLOWABLE % RTP in Table 3.7.1-1. Per Reference 7, the calculated instrument and channel uncertainties for the power range neutron flux measurement requires a further reduction of 6% RTP to assure that the maximum RTP is not exceeded with inoperable MSSVs. Therefore, when reducing the Power Range Neutron Flux-high trip setpoint, the setpoint must be reduced to less than or equal to the % RTP value shown on Table 3.7.1-1.

The allowed Completion Times are reasonable base on operating experience to complete the Required Action(s) in an orderly manner without challenging unit systems.

For each steam generator, at a specified pressure, the fractional relief capacity (FRC) of each MSSV is determined as follows:

$$FRC = \frac{A}{B}$$

where:

A—the relief capacity of the MSSV; and

B—the total relief capacity of all the MSSVs of the steam generator.

The FRC is the relief capacity necessary to address operation with reduced THERMAL POWER.

The reduced THERMAL POWER levels in the LCO prevent operation at power levels greater than the relief capacity of the remaining MSSVs. The reduced THERMAL POWER is determined as follows:

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

$$RP = 1 - \frac{(N_1 \times FRC_1 + N_2 \times FRC_2 + \dots + N_5 \times FRC_5)}{5} \times 100\%$$

where:

~~RP~~ - Reduced THERMAL POWER for the most limiting steam generator expressed as a percent of RTP;

~~N₁, N₂, ..., N₅~~ represent the status of the MSSV 1, 2, ..., 5, respectively;

~~=0~~ if the MSSV is OPERABLE;

~~=1~~ if the MSSV is inoperable;

~~FRC₁, FRC₂, ..., FRC₅~~ - the relief capacity of the MSSV 1, 2, ..., 5, respectively, as defined above.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status, THERMAL POWER for the Power Range Neutron Flux Trip is not reduced as required by Table 3.7-1 within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTSSR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4 5), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 5 6). According to Reference 5 6, the following tests are required:

- a. Visual examination;
- b. Seat tightness determination;
- c. Setpoint pressure determination (lift setting);
- d. Compliance with owner's seat tightness criteria; and
- e. Verification of the balancing device integrity on balanced valves.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a $\pm 3\%$ setpoint (as found lift point) tolerance on the valves for OPERABILITY (with the exception of the lowest set MSSV setpoint which is $(+3\%/-2\%)$; however, the valves are reset to $+1\%$ during the surveillance to allow for drift. The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section [10.3.1].
2. ~~ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.~~
3. FSAR, Section [15.2] and [15.3].
4. ~~NRC Information Notice IN-94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.~~
- 4 5. ASME, Boiler and Pressure Vessel Code, Section XI.
- 5 6. ANSI/ASME OM-1-1987, (including OM-a-1988 ADDENDA).
7. ~~Westinghouse Report WCAP-11082, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Stations Eagle 21 Version", dated May 1993.~~
8. ~~PG&E Design Calculation N-114, "Over-Pressure Study for One MSSV Per Loop Unavailable", dated 3/10/94.~~
9. ~~PG&E Design Calculation N-115, "Reduced Power Levels for A Number of MSSVs Inoperable", dated 3/14/94.~~

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are installed back to back with the MS reverse flow check valves. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by either high negative steam line pressure rate or low steam line generator pressure or high-high containment pressure. The MSIVs are held in the open position and will fail in the closed direction on loss of control air or fail open on loss of actuation power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section [10.3] (Ref. 1).

APPLICABLE
SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the FSAR, Section [6.2] 6, Appendix 6.2 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the FSAR, Section [15.1.5] 15.4.2 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for the containment pressure analysis is the SLB inside containment, with initial reactor power at 30% with a no loss of offsite power, following turbine trip, and failure of the MSIV on the affected steam generator to close. At lower powers, the steam generator inventory and temperature are

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at their maximum, maximizing the analyzed mass and energy release to the containment. Due to the assumed reverse flow (the MSIV reverse flow check valves are not credited to function even though they are Design Class I) and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIV contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available, and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an MSIV to close.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

- a. An HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is assumed to be discharged into containment from all steam generators, as no credit is taken for the MSIV reverse flow check valves until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.
- b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.

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- c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.
- d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.
- e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

This LCO requires that ~~four~~ MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated (~~vented or prevented from opening~~), when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

~~In MODE 4, normally most of the MSIVs are closed, and the steam generator energy is low, thus OPERABILITY in MODE 4 is not required.~~

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident

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occurring during this time period that would require a closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis. MSIV closure is indicated by the control room valve indicating lights or monitor light box lights.

The 8 hour Completion Time is consistent with that allowed in Condition A.

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within

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6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTSSR 3.7.2.1

This SR verifies that MSIV closure time is \leq [4-6] 5.0 seconds on an actual or simulated actuation signal. ~~The remote manual hand switch may be used as the actuation signal for this SR. The MSIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1 or 2.~~

~~The Frequency is in accordance with the [Inservice Testing Program or [18] months]. The [18] month Frequency for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

~~This test is may be conducted in MODE 3 with the unit at operating temperature and pressure, as discussed in Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated. However, the test is normally conducted in MODE 5 as permitted by the cold shutdown frequency justification provided in the Inservice Testing Program (IST) and as permitted by Reference 6, Part 10, paragraph 4.2.1.2(c).~~

SR 3.7.2.2

~~This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The frequency of MSIV testing is every 18 months. The 18 month Frequency is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.~~

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REFERENCES

1. FSAR, Section [10.3].
 2. FSAR, Section [6.2]6 Appendix 6.2.C.
 3. FSAR, Section [15.1.5] 15.4.2.
 4. 10 CFR 100.11.
 5. ASME, Boiler and Pressure Vessel Code, Section XI.
 6. ANSI/ASME OM-1-1987 (including OM-a-1988 ADDENDA)
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B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves

BASES

BACKGROUND

The safety related function of the MFRVs and the associated bypass valves is to provide the initial isolation of main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Since the MFRVs and associated bypass valves are located in non-safety related piping, the MFIVs also provide safety related isolation of the isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB) a short time later. The safety related function of the MFRVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following an HELB. Closure of the MFIVs and associated bypass valves or MFRVs and associated bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs and associated bypass valves, or MFRVs and associated bypass valves, effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

The MFIVs and associated bypass valves, or MFRVs and associated bypass valves, isolate the non-safety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MFIV and associated bypass valve, and one MFRV and its associated bypass valve, are located on each MFW line, outside but close to containment. The MFIVs and MFRVs are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MFIV or MFRV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

The MFIVs and associated bypass valves, and MFRVs and associated bypass valves, close on receipt of any safety injection (SI) signal a T - Low coincident with reactor trip (P-4) or steam generator (S76) water level-high high signal. They may also be actuated manually. The Main Feedwater Pump Turbine is also

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BASES

tripped upon receipt of an SI or S/G water level- high high signal (as well as other pump related trips), however these are Class II trips and are only credited as a backup to the single failure of a MFRV and associated bypass valve trip. The MFRVs and associated bypass valves also close on receipt of a T₁₀-Low coincident with reactor trip (P-4). In addition to the MFIVs and associated bypass valves, and the MFRVs and associated bypass valves, a check valve inside containment located upstream of the MFIV is available. The check valve isolates the feedwater line, penetrating containment, and ensures that the consequences of events do not exceed the capacity of the containment heat removal systems. Intact steam generators do not continue to feed the feedwater line break in the non-safety related piping upstream of the feedwater isolation check valves and that the AFW flow will be to the steam generators.

A description of the MFIVs and MFRVs is found in the FSAR, Section 10.4.7 (Ref. 1).

APPLICABLE
SAFETY ANALYSES

The design basis of the MFIVs and MFRVs is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs and associated bypass valves, or MFRVs and associated bypass valves, may also be relied on to terminate an SLB for core and containment response analysis and excess feedwater event upon the receipt of a steam generator water level- high high signal or a feedwater isolation signal on high-high steam generator level.

Failure of an MFIV, MFRV, or the associated bypass valves to close following an SLB or FWLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs and MFRVs satisfy Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

This LCO ensures that the MFIVs, MFRVs, and their associated bypass valves will isolate MFW flow to the steam generators, following an FWLB or main steam line break. These MFIVs valves will also isolate the non-safety related portions from the safety related portions of the system.

This LCO requires that four MFIVs and associated bypass valves and four MFRVs and associated bypass valves be OPERABLE. The MFIVs and MFRVs and their associated bypass valves are considered OPERABLE when isolation times are

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within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. If a feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, and failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

The MFIVs and MFRVs and the associated bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the blowdown of more than one steam generator. In MODES 1, 2, and 3, the MFIVs and MFRVs and the associated bypass valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs, MFRVs, and the associated bypass valves are normally closed since MFW is not required.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis

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remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated.

C.1 and C.2

With one associated bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable associated bypass valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other

(continued)

BASES

administrative controls, to ensure that these valves are closed or isolated.

D.1

With two inoperable valves in the same flow path, there may be no redundant system only the Class III main feedwater pump trip is available to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such, is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFRV, or otherwise isolate the affected flow path.

E.1 and E.2

If the MFIV(s) and MFRV(s) and the associated bypass valve(s) cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.3.1 and SR 3.7.3.2

This These SRs verifies that the closure time of each MFIV is ≤ 60 seconds and that each MFRV, and associated bypass valves is ≤ 7 seconds on an actual or simulated actuation signal. The MFIV and MFRV closure times are assumed in the accident and containment analyses. This These Surveillances is are normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code Section XI (Ref. 2) quarterly stroke requirements during operation in MODES 1 and 2.

The Frequency for this these SRs is in accordance with the / Inservice Testing Program or [18] months. The [18] month

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~~Frequency for valve closure is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency.~~

SR 3.7.3.3

This SR verifies that each MFIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MFIV testing is every 18 months. The 18 month Frequency is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.4.7.
 2. ASME, Boiler and Pressure Vessel Code, Section XI ANSI/ASME OM-1-1987 (including OM-a-1988 ADDENDA).
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B 3.7 PLANT SYSTEMS

B 3.7.4 ~~10%~~ Atmospheric Dump Valves (ADV)

BASES

BACKGROUND

The ~~10%~~ ADVs (PCV-19, PCV-20, PCV-21, and PCV-22) provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Bypass System to the condenser not be available, as discussed in the FSAR, Section ~~[10.3]~~ 15 (Ref 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST) and ~~firewater storage tank (FWST)~~. The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Dump System.

One ADV line for each of the ~~four~~ steam generators is provided. Each ADV line consists of one ADV and an associated ~~manual~~ block valve.

The ADVs are provided with upstream ~~manual~~ block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are ~~usually normally~~ provided with a ~~non-Class I~~ pressurized gas supply of ~~bottled nitrogen air~~ that, ~~on~~ with a loss of pressure in the normal instrument air supply the backup ~~non-Class I nitrogen supply~~, automatically supplies nitrogen to operate the ADVs. With the loss of both the normal air supply and the backup nitrogen supply, the normal supplies are blocked and the Class I backup air bottle system is activated. With the backup air bottle system activated, control of the valves is remote manual via the Class I control circuit from the Control Room. The nitrogen bottled air supply is sized to provide the sufficient pressurized gas to operate the ADVs for the time required for Reactor Coolant System cooldown to RHR entry conditions. In addition, ~~handwheels are provided for local manual operation~~.

~~A description of the ADVs is found in Reference 1. The ADVs are OPERABLE with only a DC power source available. In addition, handwheels are provided for local manual operation.~~

APPLICABLE
SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions ~~at the maximum allowable~~. The design rate of ~~[75]~~ 100°F per hour is applicable for two steam generators, each with one ADV. This rate is adequate to cool the unit to RHR entry conditions with

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BASES

only one steam generator and one ADV. The ADVs support the AFW cooldown function from normal zero-load temperature in the RCS to a hot leg temperature of 350°F (which is the maximum temperature allowed for placing the RHR system in service). Various cooldown rates are applicable depending upon the event and the assumed available equipment. These rates vary from a high of 100°F/hr for the SGTR event to 25°F/hr for a natural circulation cooldown event utilizing the cooling water supply available in the CST and FWST.

In the accident analysis presented in Reference 1, The ADVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accident events accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the ADVs and main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR and thus limit offsite dose is more critical than the time required to cool down to RHR conditions for this event and also for other accident events. Thus, the SGTR is the limiting event for the ADVs. The number of all four ADVs are required to be OPERABLE to satisfy the SGTR accident analysis requirements depends upon the number of unit loops and consideration of any single failure assumptions regarding the failure of one ADV to open on demand since the SGTR event assumes that the ADV on the faulted SG fails open to maximize the offsite dose and that the three intact SGs are utilized to cool the RCS at the Maximum allowable rate of 100°F/hr.

The once per 24 hour verification that backup air bottle pressure is greater than or equal to 260 psig assures that the ADVs will perform as required by the applicable safety analyses.

The ADVs are equipped with manual block valves in the event an ADV spuriously fails to open or fails to close during use.

The ADVs satisfy Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

[Three] Four ADV lines are required to be OPERABLE. One ADV line is required from each of [three] four steam generators to ensure that at least one two ADV lines is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of an second ADV line on an unaffected steam generator. The block

(continued)

BASES

valves must be OPERABLE to isolate a the failed open ADV line. A closed block valve does not renders it or its ADV line inoperable; if operator action time to open the block valve is supported in the accident analysis and the appropriate ACTION must be entered until such time that the block valve is opened.

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An ADV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand.

APPLICABILITY

In MODES 1, 2, and 3, all four ADVs are required to be OPERABLE and in MODE 4, when only the ADVs associated with the a steam generator is being relied upon for heat removal, the ADVs are required to be OPERABLE.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one required ADV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ADV lines, a non-safety grade backup in the Steam Bypass System, and MSSVs and is based on a PRA analysis and the low probability of a SGTR and LOOP event occurring during this period that would require the ADV lines. Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

B.1

With two or more ADV lines inoperable, action must be taken to restore all but one ADV line to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 72 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines.

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BASES

C.1

With three or more ADV lines inoperable, action must be taken to restore all but two ADV lines to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines.

C.1.1 and C.1.2

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within ~~18~~ hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

Plant procedures provide a 31 day verification that the 10% ADV manual block valves are open assures that the valves have not been inadvertently closed.

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened and closed either remotely or locally and throttled through their full range using the remote manual controls and the backup air bottles. This SR ensures that the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually are expected to pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.2

While not a safety function, the function of the manual block valve is to isolate a failed open ADV or isolate an ADV for repair or testing during plant operation. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement.

BASES

Operating experience has shown that these components usually are expected to pass the Surveillance when performed at the [18] month Frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.3

The function of the back-up air bottles is to assure that the ADVs will be able to be opened as required to perform a controlled cooldown of the RCS in the event of a loss of the normal air supply system. The backup air bottle system was specifically installed to allow the RCS to be cooled for a SGTR coincident with a loss of offsite power. Verification of the bottle pressure once every 24 hours allows for timely bottle replacement and trending for leaks.

REFERENCES

1. FSAR, Section [10.3] 15.
 2. WCAP 11723
 3. DCM S-25B, S-3B, AND S-4
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B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take normal suction through separate and independent valve MU-671 on the single suction lines from the condensate storage tank (CST) (LCO 3.7.6) (this valve must remain open for the applicable accident analysis assumptions to be valid) and are capable of being aligned to the firewater storage tank (FWST) (LCO 3.7.6) and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the condenser steam bypass dump valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 100% of AFW flow capacity, and the turbine driven pump provides 200% of the required capacity to the steam generators, with 100% capacity defined as the flow required to two steam generators during the AFW design basis accident analysis (loss of normal feedwater flow (Ref. 1) as assumed in the accident analysis). The pumps are equipped with independent recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds two steam generators, although each pump has the capability to be manually realigned from the control room to feed other steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The turbine driven AFW pump supplies a common header capable of feeding all steam generators with DG Vital AC powered control valves actuated to the appropriate steam generator by the Engineered Safety Feature Actuation System (ESFAS). One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the

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requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

~~The AFW System actuates automatically on steam generator water level low low by the ESFAS (LCO 3.3.2). The system also actuates on loss of offsite power, safety injection and by trip of all MFW pumps. The AFW System (both the one turbine-driven and two motor-driven AFW pumps) actuates automatically upon actuation of the anticipated transient without scram mitigating system actuation circuitry (AMSAC). The motor-driven pumps are additionally actuated by: (1) safety injection; (2) an associated bus transfer to the diesel generator signal; (3) a trip of both MFW pumps; or (4) steam generator water level low low in one of four SGs. The turbine-driven pump is additionally actuated by 12 KV bus undervoltage or steam generator low low level in two of four SGs via ESFAS (LCO 3.3.2).~~

The AFW System is discussed in the FSAR, Section [10.4.9] 6.5 (Ref. 1).

APPLICABLE
SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the at least two steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3% tolerance plus 3% accumulation within 1 minute after event initiation.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and AFW spillage through feedwater line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

- a. Feedwater or Main Steam Line Break (FWLB); and

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BASES

- b. Loss of MFW (the coincident loss of offsite power is a less limiting transient since RGP heat input is lost).

In addition, the minimum available AFW flow and system characteristics are serious considerations must be considered in the analysis of normal cooldown and of a small break loss of coolant accident (LOCA) due to their potential impact.

The AFW System is also designed for decay heat removal following a Steam Generator Tube Rupture (SGTR). As such the steam turbine driven AFW pump has redundant steam supplies to assure continued availability following a SGTR or MSIB event.

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment on loss of MFW, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. In such a case, the ESFAS logic may not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly. One motor driven AFW pump would deliver to the broken MFW header at the pump runout maximum flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine driven pump and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power. DC Vital AC power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. [Three] independent AFW pumps in [three] diverse trains are required to be OPERABLE to ensure the availability of RHR decay and residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses. The and having the third AFW pump is powered by a different means, a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs. To assure steam turbine driven AFW pump operability via redundant steam supplies, steam traps 104, 105 and 106 on the supply lines must be operable or bypassed to

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BASES

ensure adequate condensate removal and check valves MS-6166 and MS-6167 must be operable

The AFW System supply is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps, each powered by a separated vital bus, be OPERABLE in two diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

The operability of the AFW suction flow path is assured by verifying the condensate storage tank outlet valve open and by verifying the capability to align the fire water storage tank to the AFW pump suction.

The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

ACTIONS

A.1

If one of the two steam supplies to the turbine driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

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BASES

- a. The redundant OPERABLE steam supply to the turbine driven AFW pump;
- b. The availability of redundant OPERABLE motor driven AFW pumps; and
- c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

If Condition A, an inoperable steam supply to the turbine driven AFW pump, is entered while, for instance, motor driven AFW pump 1-2 is inoperable and the motor driven AFW pump 1-2 is subsequently returned to an OPERABLE condition shortly after Condition A is entered, the LCO may already have not been met for up to 72 hours. This could lead to a total of up to 10 days for restoration of the motor driven AFW pump 1-2 and the turbine driven AFW pump steam supply. If before the steam supply is returned OPERABLE motor driven AFW pump 1-3 becomes inoperable, the AFW system could be inoperable for as long as 13 days.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the

(continued)

BASES

LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

D.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with non-safety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

E.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the

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BASES

inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

F.1

With the CST supply to the AFW pump suction unavailable, the primary safety related source of water required for cooldown is unavailable. The four hours required to restore the CST supply to the AFW pumps is a reasonable time to limit the risk from an event requiring the plant to cooldown.

G.1

With the FWST supply incapable of alignment to the AFW suction the required additional source of water needed for a natural circulation cooldown is unavailable. The seven days required to restore the ability for FWST realignment is a reasonable time to limit the risk from an event requiring a natural circulation cooldown based upon the available non-Class I AFW sources.

H.1 and H.2

When the Required Action F.1 or G.1 cannot be completed within the Required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 without reliance upon the steam generator for heat removal within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

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BASES

The 31 day Frequency ~~is~~ based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by ~~Section XI~~ of the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code, ~~Section XI~~ (Ref. 2) (only required at 3 month intervals) satisfies this requirement. ~~The [31] day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.~~

This SR is modified by a Note indicating that the SR ~~for the turbine-driven AFW pump~~ should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

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SURVEILLANCE
REQUIREMENTS
(continued)SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states the SR is not required in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the required AFW train may already be aligned and operating.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR for the turbine-driven pump can be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. [The] Note 2 states that the SR is not required in MODE 4. In MODE 4, the required motor-driven pump is already operating and the autostart function is not required. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.

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BASES

SURVEILLANCE
REQUIREMENTS

~~SR 3.7.5.4 (continued)~~

~~Reviewer's Note: Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.~~

~~SR 3.7.5.5 NOT USED~~

~~This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned. (This SR is not required by those units that use AFW for normal startup and shutdown.)~~

~~SR 3.7.5.6~~

~~This SR verifies that the FWST is capable of being aligned to the AFW pump suction. This assures that this additional supply of required AFW is available from the seismically qualified FWST should it be needed for a natural circulation cooldown.~~

~~The 92 day Frequency, based on engineering judgement, is consistent with procedural controls governing valve operation, and ensures correct valve positions.~~

~~A similar SR is not required for the CST alignment since the AFW system is used for startup and an AFW pump is tested each month. This operation and the pump tests assure proper valve alignment.~~

REFERENCES

1. FSAR, Section [10.4.9] ~~6.5~~ and section ~~15.2.8~~.
2. ASME, Boiler and Pressure Vessel Code, Section XI ~~ANSI/ASME OM-1-1987~~ (including ~~OMA-1988~~ ADDENDA).
3. ~~DCM S-3B~~

B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST) and Feed Water Storage Tank (FWST)

BASES

BACKGROUND

The CST supplemented by the FWST provides a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST and FWST provides a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric dump valves if the main steam isolation valves are closed. The AFW pumps operate with a continuous recirculation to the CST.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the steam-bypass condenser dump valves. The condensed steam is returned to the CST by the condensate transfer pumps. This has the advantage of conserving condensate while minimizing releases to the environment.

Because the CST and FWST are the principal components in for removing residual heat from the RCS, it is they are designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The CST and FWST are designed to Seismic Category I to ensure availability of the feedwater supply. Feedwater is also available from alternate sources as described in the FSAR.

A description of the CST is found in the FSAR, Section 9.2.6 (Ref. 1).

APPLICABLE
SAFETY ANALYSES

The CST and FWST provides cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the FSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs, followed by a cooldown to residual heat removal (RHR) entry conditions at the design cooldown rate. The limiting event for AFW supply, i.e., CST and FWST minimum volumes, is based on a loss of offsite power which assumes a reduced Reactor Coolant System (RCS) cooldown rate and requires seismically qualified water sources. The lower RCS cooldown rate on natural circulation increases the cooldown period until the residual heat removal (RHR) system can be used to remove further decay heat. The extended cooldown time thus

(continued)

BASES

~~requires more AFW supply than can be provided by the seismically qualified portion of the CST~~

~~The limiting event for the~~ Other events requiring condensate volume is ~~are~~

- 1) ~~the large feedwater line break coincident with a loss of offsite power. Single failures that also affect this event include the following:~~
 - a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine); and
 - b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

~~A nonlimiting event considered in CST inventory determinations is and:~~

- 2) ~~a break in either the main feedwater or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, since the Emergency Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.~~

The CST satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

To satisfy ~~accident Hosgr~~ analysis assumptions, the CST and FWST must contain sufficient cooling water to remove decay heat for ~~[30-60 minutes]~~ following a reactor trip from 102% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine, ~~or before isolating AFW to a broken line.~~

The CST level required is equivalent to a usable volume of ~~> [110,000 gallons]~~ 41.3% indicated level (164,678 gallons), which is ~~The FWST level required is equivalent to a usable volume~~

(continued)

BASES

of $\geq 41.75\%$ indicated level (115,844 gallons) for two units operating and $\geq 22.2\%$ indicated level (57,922 gallons) for one unit operating. These levels are based on holding the unit in MODE 3 for 2 hours, followed by a natural circulation cooldown to RHR entry conditions at $[75] 25^\circ\text{F}/\text{hour}$. This basis is established in Reference 4 and exceeds the volume required by the accident analysis.

The OPERABILITY of the CST and FWST is determined by maintaining the tank levels at or above the minimum required levels.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST is and FWST are required to be OPERABLE.

In MODE 5 or 6, the CST is or FWST are not required because the AFW System is not required.

ACTIONS

A.1 and B.1

If the CST level is not within limits, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours and once every 12 hours thereafter. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the AFW pumps are OPERABLE, and that the backup supply has the required volume of water available. The CST must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CST. If the CST or FWST levels are not within limits, the CST level must be restored within 4 hours and the FWST level must be restored within 7 days to return the AFW sources to an OPERABLE status. The CST must be restored to OPERABLE status within 4 hours because it is the primary Class 1 AFW supply. The 4 hour Completion Time provides time to restore the required CST level from the condenser or other source, and is a reasonable time to limit the risk from accidents or events requiring the plant to cooldown. The 7 day Completion Time provides time to restore the required FWST level is a reasonable time to limit the risk of a natural circulation cooldown event that would require the use of the backup volume in addition to the volume contained in the CST. Alternate non-

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BASES (continued)

seismically qualified water sources are also available to supply water to supplement the CST volume.

C.1 and C.2

If the CST or FWST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTSSR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. ~~(The required CST volume may be single value or a function of RCS conditions.)~~ The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST levels.

SR 3.7.6.2

This SR verifies that the FWST contains the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the FWST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the FWST levels.

REFERENCES

1. FSAR, Section 9.2.6 and 9.5.1.
2. FSAR, Chapter 6.
3. FSAR, Chapter 15.

4. DCM S-3B

B 3.7 PLANT SYSTEMS

B 3.7.7 ~~Vital~~ Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water Auxiliary Saltwater (ASW) System, and thus to the environment.

~~A typical CCW System is arranged as two independent, full capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. The CCW system consists of three CCW pumps powered by separate vital buses, two CCW heat exchangers, and an internally baffled two chamber CCW surge tank. The piping system consists of three parallel loops headered out of the two parallel 100% capacity heat exchangers. Two loops are separable redundant full capacity vital service loops which serve only ESF equipment and the non-redundant post-LOCA sample cooler. The vital loops are separable to mitigate a single passive failure during post-LOCA long term cooling. A third miscellaneous service loop serves nonvital equipment. The divided surge tank is connected to the vital header return piping and is sized to meet system leakage requirements and maintain adequate NPSH on system pumps.~~

~~The CCW system is hydraulically balanced to ensure that sufficient cooling water is delivered to ESF loads on the vital loops during a DBA.~~

~~The CCW system is designed to perform its function with a single failure of any component. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The All three pumps are in each train is automatically started on receipt of a safety injection signal, and all nonessential components are isolated. the miscellaneous service loop is automatically isolated on hi-hi containment pressure.~~

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section 9.2.2 (Ref. 1). The principal safety related function of the CCW System is the removal of accident generated containment heat via the containment fan cooling units (CFCUs).

(continued)

BASES

and removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. This Decay heat removal may be during a normal or post accident cooldown and shutdown.

APPLICABLE
SAFETY ANALYSES

The design basis of the CCW System is for one CCW train loop to remove the post loss of coolant accident (LOCA) DBA heat load from the containment sump during the recirculation phase, without exceeding the design basis continuous a maximum CCW temperature of [120]°F (Ref. 2). The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA each model the maximum and minimum performance of the CCW System, respectively. The normal temperature of the CCW is [80]°F, and, during unit cooldown to MODE 5 ($T_{\text{cold}} < [200]^\circ\text{F}$), a maximum temperature of 95°F is assumed. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the ECCS pumps, 132°F and not to exceed 120°F with an allowable transient not to exceed 140°F for more than 6 hours (Ref. 1).

In accordance with GDC 44, the CCW system is designed to provide sufficient heat removal for normal and post accident ESF heat loads without overheating. The CCW system and ASW system are essentially considered a single heat removal system for the purpose of assessing the ability to sustain either a single active or passive failure and still perform design basis heat removal. Only one ASW pump and one CCW heat exchanger is required, as assumed in the safety analysis, to provide sufficient heat removal from containment to mitigate a DBA. However, to ensure maximum heat removal capability, operators are instructed to place the second CCW heat exchanger in service early in the emergency operating procedures.

The CCW System also functions to cool the unit from RHR entry conditions ($T_{\text{coldave}} < 350^\circ\text{F}$), to MODE 5 ($T_{\text{coldave}} < 200^\circ\text{F}$), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW heat exchangers and RHR trains heat exchangers operating. One CCW train exchanger is sufficient to remove decay heat during subsequent operations with $T_{\text{cold}} < [200]^\circ\text{F}$. This assumes a maximum service water temperature of [95]°F occurring simultaneously with the maximum heat loads on the system.

In the event that CCW system leakage occurs and system makeup is not available, the surge tank volume provides a minimum of 20 minutes, based on a non-mechanistic leakage rate of 200 gpm, for operators to locate and isolate the leak or realign the CCW

(continued)

BASES

system into two separate loops before the system becomes impaired due to water loss.

The CCW System satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

In the event of a DBA, one CCW loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two vital loops of CCW must be OPERABLE. At least one CCW loop will operate assuming the worst case single active failure occurs coincident with a loss of offsite power. To meet the LCO on Component Cooling Water loops, vital headers A and B, both CCW heat exchangers, the surge tank, and all three CCW pumps must be operable.

A vital CCW loop is considered OPERABLE when:

- a. Two CCW pumps, one CCW heat exchanger, and the surge tank are OPERABLE; and
- b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System except for isolation of CCW to the CFCUS. Isolation of CCW to the CFCUS could potentially affect the flow balance and requires evaluation to ensure continued operability.

Split loop alignment of the CCW system during normal operation requires Condition A to be entered because the CCW system cannot tolerate a single failure in this configuration.

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its principal safety related function of removal of accident generated containment heat via the CFCUS and removal of decay heat from the reactor via the Residual Heat Removal (RHR) System.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

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BASES

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable vital CCW train loop results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one vital CCW train loop is inoperable, action must be taken to restore two vital CCW loops to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE vital CCW train loop is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train overall heat transfer capability of ultimate heat sink system, operator action, and the low probability of a DBA occurring during this period.

Split loop alignment of the CCW system during normal operation requires Condition A to be entered because the CCW system cannot tolerate a single failure in this configuration.

B.1 and B.2

If the vital CCW train loop cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW System. A possible exception to this note is isolation of CCW to the CFCUs. Isolation of CCW to the CFCUs could potentially affect the flow balance and requires evaluation to ensure continued operability.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the

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proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated ~~safety related~~ actuation signal. ~~The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation.~~ The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. ~~This surveillance requirement applies to the SIS auto-start and the 4KV auto-transfer automatic starts only.~~ Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section ~~9.2.2~~.
 2. FSAR, Section ~~6.2~~.
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BASES

B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water Auxiliary Saltwater System (SWS) (ASW)

BASES

BACKGROUND

The SWS ASW provides a heat sink from the Pacific Ocean for the removal of process and operating heat from safety related components via the component cooling water (CCW) system during all modes of operation including a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the SWS ASW also provides this function for various safety related and non-safety related components. The safety related function is covered by this LCO.

The SWS ASW consists of two separate, 100% capacity, safety related, cooling water trains. Each train consists of two one 100% capacity pump, one component cooling water (CCW) heat exchanger, piping, valving, and instrumentation, and two cyclone separators. The pumps and valves are remote and manually aligned, except in the unlikely event of a loss of coolant accident (LOCA). The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal or 4KV automatic transfer, and all essential valves are aligned to their post accident positions. The SWS also provides emergency makeup to the spent fuel pool and CCW System [and is the backup water supply to the Auxiliary Feedwater System]. Normal configuration is for one train operation with the second pump cross-tied in stand-by and the second heat exchanger valved out-of-service except when the UHS temperature is 64°F or higher therefore no valve realignment occurs with a safety injection signal. Manual and remote manual system realignment provides for utilization of the second CCW heat exchanger, for use of the standby pump on the same unit, for cross-tying the standby ASW pump from opposite unit, and for train separation for long term cooling. The ASW unit cross-tie valve (FCV-601) allows one ASW pump on one unit to supply the CCW heat exchanger(s) on the other unit. FCV-601 is controlled by ECG 17.1.

Additional information about the design and operation of the SWS ASW system, along with a list of the components served, is presented in the FSAR, Section [9-2-1] 9-2-7 (Ref. 1). The principal safety related function of the SWS ASW system is the removal of decay heat from the reactor via the vital CCW System.

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BASES

APPLICABLE
SAFETY ANALYSES

The design basis of the SWS ASW system is for one SWS ASW train, in conjunction with the CCW System and a 100% capacity the containment cooling systems, to remove accident generated and core decay heat following a design basis LOCA as discussed in the FSAR, Section [6.2] (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System by the ECCS pumps. The ASW system can be re-configured to maintain the CCW temperature to within its design bases limits. The SWS ASW system is designed to perform its function with a single failure of any active component, assuming with or without the loss of offsite power. This assumes a maximum SWS ASW temperature of [95] 64°F occurring simultaneously with maximum heat loads on the system.

The SWS ASW system, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR), as discussed in the FSAR, Section [5.4.7], (Ref. 3) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of ASW pumps, CCW heat exchangers, and RHR System trains/heat exchangers that are operating. One SWS train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6.

The SWS ASW system satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

Two SWS ASW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An SWS ASW train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

- a. The pump is OPERABLE; and
- b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE capable of performing their intended safety functions. The standby cross-tied ASW pump discharge (via FCV 495 and 496) provides redundancy for the operating ASW pump. The standby heat exchanger, valved out for operating convenience, is available to provide additional heat removal capability by valving in the second heat exchanger. Valving is provided to permit train

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BASES

separation during the long term cooling phase of a LOCA should a single passive failure occur.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS ASW system is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SWS ASW system and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the SWS ASW system are determined by the systems it supports.

ACTIONS

A.1

If one SWS ASW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SWS ASW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SWS ASW train could result in loss of SWS ASW system function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources Operating," should be entered if an inoperable SWS train results in an inoperable emergency diesel generator. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," should be entered if an inoperable SWS ASW train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the SWS ASW train cannot be restored to OPERABLE status within the associated Completion Times, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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BASES

SURVEILLANCE
REQUIREMENTSSR 3.7.8.1

~~This SR is modified by a Note indicating that the isolation of the SWS components or systems may render those components inoperable, but does not affect the OPERABILITY of the SWS.~~

Verifying the correct alignment for manual, and power operated, and automatic valves in the SWS ASW system flow path provides assurance that the proper flow paths exist for SWS ASW system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position and that those valves requiring remote positioning have available power and air supplies such that if required, the valve would be capable of being placed in its required position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper automatic remote manual full stroke operation of the SWS ASW valves on an actual or simulated actuation signal. ~~The SWS is a normally operating system that cannot be fully actuated as part of normal testing.~~ This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month 92 day Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. ~~engineering judgement is consistent with procedural controls governing valve operation~~ the IST program frequency and is consistent with the ASME O&M Code testing requirements, and ensures the ability to correctly align the valves. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month 92 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

(continued)

BASES

SR 3.7.8.3

This SR verifies proper automatic operation of the SWS ASW pumps on an actual or simulated safety related actuation signal. The SWS ASW is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This surveillance requirement applies to the SIS auto-start and the 4kV auto transfer automatic starts only. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section ~~[9.2.1]~~ 9.2.7.
2. FSAR, Section 6.2.
3. ~~FSAR, Section [5.4.7].~~ NRC Generic Letter 91-13, "Request for Information Related to the Resolution of Generic Issue 130, Essential Service Water System Failures at Multi-unit Sites, Pursuant to 10 CFR 50.54 (F)," dated September 19, 1991.

B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for processing and operating transferring heat from safety related components during a transient or accident, as well as safety related and non-safety related heat loads during normal operation. This is done by utilizing the service Water Pacific Ocean, the Auxiliary Saltwater System (ASW) and the Component Cooling Water (CCW) System.

The UHS is common to both units and has been defined as that complex of water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), the Pacific Ocean and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as discussed in the FSAR, Section [9.2.5] (Ref. 1). If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink. The two principal functions of the UHS are dissipation of heat during normal operation, the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

A variety of complexes is used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required.

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. Basins of cooling towers generally include less than a 30 day supply of water, typically 7 days or less. A 30 day supply would be dependent on other source(s) and makeup system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a 1 day supply, the systems for replenishing the basin and the backup source(s) become of sufficient importance that the makeup system itself may be required to meet the same design criteria as an Engineered Safety Feature (e.g., single failure considerations), and multiple makeup water sources may be required. To ensure UHS availability, ASW components located within the projected sea wave zone are designed to operate during extreme ocean levels for a short duration (for example, tsunami run up and draw down conditions) per Reference 2. To maintain adequate cooling for safety related equipment, operational limits are established based on ocean supply temperature per Reference 4.

BASES

Additional information on the design and operation of the system along with a list of components served, can be found in Reference 1.

APPLICABLE
SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core and containment following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. For units that use UHS as the normal heat sink for condenser cooling via the Circulating Water System, unit operation at full power is its maximum heat load. Its maximum post accident heat load occurs 20 minutes after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation and the containment cooling systems and RHR are required to remove the core decay heat. The Pacific Ocean as a single water source for the Ultimate Heat Sink will provide in excess of 30 days of cooling water during normal and emergency shutdown conditions as required by AEC Safety Guide 27 (Ref. 3).

The operating limits are based on conservative heat transfer analyses for the worst case LOCA. Reference 1 provides the details of the assumptions used in the analysis, which include worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and worst case single active failure (e.g., single failure of a manmade structure). The UHS is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water in the UHS.

The UHS satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it contains a sufficient volume of water is at or below the maximum temperature that would allow the SWS ASW to operate for at least 30 days following the design basis LOCA without the loss of net positive suction head (NPSH), and DBA without exceeding the maximum design temperature of the equipment served by the SWS CCW system. To meet this condition, the UHS temperature should not exceed [90 F] 64°F unless two CCW heat exchangers are in service and the level should not fall below [562 ft mean sea level] during normal unit operation.

APPLICABILITY

In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

BASES

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1

~~If one or more cooling towers have one fan the UHS is inoperable (i.e., up to one fan per cooling tower inoperable inlet water temperature > 64°F), action must be taken to restore the inoperable cooling tower fan(s) UHS to OPERABLE status within 7 days 8 hours by placing a second CCW heat exchanger in service must be performed within 8 hours. This action provides assurance that the ASW system and the CCW system can operate within its temperature limit.~~

The 7-day 8 hour Completion Time is reasonable based on the low probability of an accident occurring during the 7 days 8 hours that one cooling tower fan is inoperable (in one or more cooling towers), the number of available systems, the temperature is > 64°F without two CCW heat exchangers in service and the time required to reasonably complete the Required Action.

B.1 and B.2

~~If the cooling tower fan the second heat exchanger cannot be restored to OPERABLE status placed in service within the associated Completion Time or] if the UHS is inoperable for reasons other than Condition A, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.~~

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.9.1 ~~Not used~~

~~This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the SWS pumps. The [24] hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is [562] ft [mean sea level].~~

BASESSR 3.7.9.2

~~This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the UHS is [90 F].~~

~~This SR verifies that adequate long term (30 day) cooling can be maintained. The 24, 12 and 2 hour surveillance Frequencies are based on operating experience related to trending of the temperature variations during the applicable MODES. This SR verifies the temperature of the UHS so that appropriate actions can be taken to assure that the ASW system can continue to assure that the CCW system will not exceed its design temperature profile.~~

SR 3.7.9.3 Not Used

~~Operating each cooling tower fan for > [15] minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.~~

SR 3.7.9.4 Not Used

~~This SR verifies that each cooling tower fan starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with the typical refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

REFERENCES

1. FSAR, Section [9.2.5].
2. Regulatory Guide 1.27. ~~FSAR Sections 2.4.11.5 & 2.4.11.6~~
3. ~~AEC Safety Guide 27~~
4. ~~SSER 16~~

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Emergency Filtration Ventilation System (CREEVS)

BASES

BACKGROUND

The CREEVS provides a protected environment from which operators can control the units from the common control room following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The CREEVS consists of two independent, redundant trains that recirculate and filter the control room air (one train from each unit). Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a one pressurization supply fan, one filter booster fan, and one main supply fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters to remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank.

The CREEVS is an emergency system, parts of which may also operate during normal unit operations, in the standby mode of operation. Upon receipt of the an actuating signal(s), the normal air supply to the control room is isolated, and the stream of outside ventilation air from the pressurization system and is recirculated control room air is passed through the system filter trains. The pressurization system draws outside air from either the north end or the south end of the turbine building based upon the wind direction or the absence of releases at the inlet. The prefilters or demisters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each filter train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. Both the demister and heater is are important to the effectiveness of the charcoal adsorbers.

Manual or automatic actuation of the CREEVS places the system in one of three either of two separate states: 1) pressurization (MODE 4), 2) recirculation (MODE 3), or 3) smoke removal (MODE 2) (emergency radiation state or toxic gas isolation state) of the emergency mode of operation, depending on the initiation signal. Actuation of the system to the recirculation mode emergency radiation state of the emergency mode of operation, closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the control room air through the redundant trains of HEPA and the charcoal filters. The pressurization mode emergency radiation state also

(continued)

BASES

initiates pressurization and filtered ventilation of the air supply to the control room.

Outside air is filtered, diluted via pressure equalization with building air from the mechanical electrical equipment and cable spreading rooms, and added to the air being recirculated from the control room. Pressurization of the control room prevents infiltration of unfiltered air from the surrounding areas of the building. The actions taken in the manual actuation of the recirculation mode toxic gas isolation state are the same, except that the signal switches control room ventilation to an isolation alignment to prevent outside air from entering the control room.

To monitor the status of the booster fan(s) small plastic streamers are installed on the exhaust duct grates. These exhaust ducts are located in the back of the control room in the ceiling and are used to take suction on the control room atmosphere. These streamers will hang down when the booster fan(s) are not operating. Therefore if a booster fan is in operation the streamers will be up. This will permit the operators to diagnose a problem with the booster fan or with the booster fan supply damper.

The pressurization mode is the only automatically actuated mode change since bulk chlorine gas is no longer kept onsite and the chlorine monitors which previously initiated the recirculation mode have been deactivated.

The air entering the control room is continuously monitored by radiation and toxic gas detectors. One detector output above the setpoint will cause actuation of the pressurization mode emergency radiation state or toxic gas isolation state, as required. The actions of the toxic gas isolation state are more restrictive, and will override the actions of the emergency radiation state.

A single train will pressurize the control room equal to about or greater than 0.125 inches water gauge. The CREEVS operation in maintaining the control room habitable is discussed in the FSAR, Section 9.4.1 [6.4] (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREEVS is designed in accordance with Seismic Category I requirements.

The CREEVS is designed to maintain the control room environment for the duration of the most severe 30 days of continuous

(continued)

BASES

occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE
SAFETY ANALYSES

The CREEVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the control room envelope ensures an adequate supply of filtered air to all areas requiring access. The CREEVS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in the FSAR, Chapter 15 (Ref. 2).

The analysis of toxic gas releases demonstrates that the toxicity limits are not exceeded in the control room following a toxic chemical release, as presented in Reference 1.

The worst case single active failure of a component of the CREEVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CREEVS satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

Two independent and redundant CREEVS trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. The redundant train means a second train from the other unit (Ref. 5). Total system failure could result in exceeding a dose of 5 rem to the control room operator in the event of a large radioactive release.

The CREEVS is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CREEVS train is OPERABLE when the associated:

- a. main supply fan (one), filter booster fan (one) and pressurization fan (one) Fan are ~~is~~ OPERABLE;
- b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and
- c. Heaters, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, 3, 4, [5, and 6,] and during movement of irradiated fuel assemblies [and during CORE ALTERATIONS], CREEVS must be OPERABLE to control operator exposure during and following a DBA or the the CREEVS is required to cope with the release from the rupture of an outside waste gas tank.

During movement of irradiated fuel assemblies [and CORE ALTERATIONS], the CREEVS must be OPERABLE to cope with the release from a fuel handling accident.

~~CRVS OPERABILITY requires that for MODE 5 and 6 and during movement of irradiated fuel assemblies in either unit, when there is only one OPERABLE train of CRVS, the OPERABLE CRVS train must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which is energizing the OPERABLE CRVS train. This is an exception to LCO 3.0.6.~~

ACTIONS

A.1

When one CREEVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREEVS train is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREEVS train could result in loss of CREEVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

~~CRVS OPERABILITY requires that for MODE 5 and 6 and during movement of irradiated fuel assemblies, when there is only one OPERABLE train of CRVS, the OPERABLE CRVS train must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which is energizing the OPERABLE CRVS train. This is an exception to LCO 3.0.6.~~

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CREEVS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

C.1, C.2.1, and C.2.2

In ~~MODE 5 or 6, or~~ during movement of irradiated fuel assemblies [~~or during CORE ALTERATIONS~~], if the inoperable CREEVS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CREEVS train in the recirculation emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected. As noted above, if only one CRVS train is OPERABLE, the OPERABLE train must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which is energizing the OPERABLE CRVS train. The power requirements for the one OPERABLE CRVS train assures that the ventilation function will not be lost during a fuel handling accident with a subsequent loss of off-site power.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

~~Required Action C.1 is modified by a Note indicating to place the system in the toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable.~~

D.1 and D.2

In ~~MODE 5 or 6, or~~ during movement of irradiated fuel assemblies [~~or during CORE ALTERATIONS~~], with two CREEVS trains inoperable, action must be taken immediately to suspend activities, including positive reactivity changes, that could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREEVS trains are inoperable in MODE 1, 2, 3, or 4, the CREEVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

Once actuated due to a fuel handling accident the CRVS must be protected against a single failure. This protection, although not required for immediate accident response, is assured by requiring that a backup power supply be provided as described above in Applicability. This back up is assured via the performance of non-TS surveillances that verify the ability to transfer power supplies.

The 31 day procedural verification of the separate vital power supplies for the redundant fans and the one hour operation of each supply, booster and pressurization supply fan (unless already operating) assures system reliability and two train redundancy.

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized and operating automatically (filter temperature control)]. ~~Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.~~ The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.10.2

This SR verifies that the required CREEVS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The CREEVS filter tests are in accordance with Regulatory Guide 1.52 ANSI 510-1980 (Ref. 3). The [VFTP] includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.10.3

This SR verifies that each CREEVS train automatically starts and operates in the pressurization mode on an actual or simulated actuation signal generated from a Phase A Isolation. The Frequency of [18] months is specified in Regulatory Guide 1.52 ANSI 510-1980 (Ref. 3).

(continued)

BASES

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CREEVS. During the ~~pressurization~~ emergency mode of operation, the CREEVS is designed to pressurize the control room \geq ~~{0.125}~~ inches water gauge positive pressure with respect to ~~the outside atmosphere and~~ adjacent areas in order to prevent unfiltered inleakage. The CREEVS is designed to maintain this positive pressure with one train ~~at a makeup flow rate of [3000] cfm.~~ The Frequency of ~~{18}~~ months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).

REFERENCES

1. FSAR, Section ~~9.4.1~~ [6.4].
 2. FSAR, Chapter ~~15~~.
 3. Regulatory Guide ~~1.52~~, Rev. 2 ~~ANSI 510-1980~~.
 4. NUREG-0800, Section 6.4, Rev. 2, July 1981.
 5. ~~DCM S-23F~~
-

B 3.7 PLANT SYSTEMS

B 3.7.11 ~~Not Used~~ Control Room Emergency Air Temperature Control System (CREATCS)

BASES

BACKGROUND

~~The CREATCS provides temperature control for the control room following isolation of the control room.~~

~~The CREATCS consists of two independent and redundant trains that provide cooling and heating of recirculated control room air. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for control room temperature control. The CREATCS is a subsystem providing air temperature control for the control room.~~

~~The CREATCS is an emergency system, parts of which may also operate during normal unit operations. A single train will provide the required temperature control to maintain the control room between [70]°F and [85]°F. The CREATCS operation in maintaining the control room temperature is discussed in the FSAR, Section [6.4](Ref. 1).~~

APPLICABLE
SAFETY ANALYSES

~~The design basis of the CREATCS is to maintain the control room temperature for 30 days of continuous occupancy.~~

~~The CREATCS components are arranged in redundant, safety related trains. During emergency operation, the CREATCS maintains the temperature between [70]°F and [85]°F. A single active failure of a component of the CREATCS, with a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The CREATCS is designed in accordance with Seismic Category I requirements. The CREATCS is capable of removing sensible and latent heat loads from the control room, which include consideration of equipment heat loads and personnel occupancy requirements, to ensure equipment OPERABILITY.~~

~~The CREATCS satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).~~

(continued)

BASES (continued)

LCO

~~Two independent and redundant trains of the CREATCS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.~~

~~The CREATCS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the heating and cooling coils and associated temperature control instrumentation. In addition, the CREATCS must be operable to the extent that air circulation can be maintained.~~

APPLICABILITY

~~In MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies [and during CORE ALTERATIONS], the CREATCS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.~~

~~[In MODE 5 or 6,] CREATCS may not be required for those facilities that do not require automatic control room isolation.~~

ACTIONS

A-1

~~With one CREATCS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREATCS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CREATCS train could result in loss of CREATCS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or nonsafety related cooling means are available.~~

(continued)

BASES

ACTIONS
(continued)

B.1 and B.2

~~In MODE 1, 2, 3, or 4, if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.~~

C.1, C.2.1, and C.2.2

~~[In MODE 5 or 6, or] during movement of irradiated fuel [or during CORE ALTERATIONS], if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREATCS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected.~~

~~An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.~~

D.1 and D.2

~~[In MODE 5 or 6, or] during movement of irradiated fuel assemblies [or during CORE ALTERATIONS], with two CREATCS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.~~

ACTIONS
(continued)

E.1

~~If both CREATCS trains are inoperable in MODE 1, 2, 3, or 4, the control room CREATCS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.~~

SURVEILLANCE
REQUIREMENTS

SR 3.7.11.1

~~This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the [safety analyses] in~~

(continued)

BASES

~~the control room. This SR consists of a combination of testing and calculations. The 18 month Frequency is appropriate since significant degradation of the CREATCS is slow and is not expected over this time period.~~

REFERENCES

- ~~1. FSAR, Section [6.4].~~
-

B 3.7 PLANT SYSTEMS

B 3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS) Auxiliary Building Ventilation System (ABVS)

BASES

BACKGROUND

The ECCS PREACS ABVS filters air from the area of the active ECCS components during the recirculation phase of a loss of coolant accident (LOCA). The ECCS PREACS ABVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area. If one of the pumps is operating and the lower reaches of the auxiliary building.

The ECCS PREACS ABVS consists of two independent and redundant trains. Each train is powered by a separate vital bus and consists of a supply fan and an exhaust fan. A single roughing and HEPA filter is common to both trains for normal operations and a single roughing filter HEPA filter and charcoal adsorber bank and a single manually initiated heater are common to both trains for emergency operations. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the pump room following receipt of a safety injection (SI) signal.

The ABVS has several modes of operation. These modes include: (1) Building Only; (2) Building and Safeguards; and (3) Safeguards only. In the Building Only mode of operation, the ABVS provides ventilation flow to all parts of the auxiliary building except for the ECCS pump rooms, but does take suction from the ECCS rooms. If any ECCS pump is started, the ABVS will automatically re-align to the Building and Safeguards mode of operation. In the Building and Safeguards mode of operation, ventilation is provided to the entire auxiliary building, including the ECCS pump rooms. In the Safeguards Only mode of operation, only the ECCS pump rooms and the lower reaches of the auxiliary building are provided with ventilation. If a SI signal is generated, the system will automatically realign such that all exhaust flow from the ECCS pump rooms passes through the common HEPA filter/charcoal adsorber bank prior to being exhausted to

(continued)

BASES

atmosphere. Whenever an SI signal is generated, the operator must manually energize the heater from the control room.

The ECCS PREACS is a standby system, aligned to bypass the system HEPA filters and charcoal adsorbers. During emergency operations, the ECCS PREACS dampers are realigned, and fans are started to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the ECCS pump room isolate, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ECCS PREACS ABVS is discussed in the FSAR, Sections [6.5.1], [9.4.5] 2, and [15.6.5] 5 (Refs. 1, and 2, and 3, respectively) since it may be used for normal, as well as post accident, ventilation and atmospheric cleanup functions. The primary purpose of the single manually initiated heaters is to maintain the relative humidity at an acceptable level, consistent with iodine removal efficiencies per Regulatory Guide 1.62 ASTM D 3803-1989 (Ref. 4 3). There is no redundant heater since the failure of the charcoal absorber and heater train would constitute a second failure in addition to the RHR pump seal failure assumed in conjunction with a LOCA (Ref 7).

APPLICABLE
SAFETY ANALYSES

The design basis of the ECCS PREACS ABVS is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as an SI RHR pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 CFR 100 (Ref. 5) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of Reference 5 limits). The analysis of the effects and consequences of a large break LOCA is presented in Reference 3 2. The ECCS PREACS ABVS also functions, following a small break LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing.

The ventilation flow is also required to maintain the temperatures of the operating ECCS motors within allowable limits. The ventilation function has been designed for single failure and the system will continue to function to provide its ECCS motor cooling function.

Two types of system failures are considered in the accident analysis for radiation release: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower

(continued)

BASES

efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

The ~~ECCS PREACS~~ ABVS satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

Two independent and redundant trains of the ~~ECCS PREACS~~ ABVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. Total system failure could result in the atmospheric release from the ECCS pump room exceeding 10 CFR 100 limits in the event of a Design Basis Accident (DBA).

~~ECCS PREACS~~ ABVS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration and temperature are OPERABLE in both trains.

An ~~ECCS PREACS~~ ABVS train is considered OPERABLE when its associated:

- a. ~~Supply and exhaust fans~~ are OPERABLE;
 - b. ~~The common roughing filter~~ HEPA filter and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and
 - c. ~~A heater, demister, ductwork, valves, and dampers~~ are OPERABLE and air circulation can be maintained.
-

APPLICABILITY

In MODES 1, 2, 3, and 4, the ~~ECCS PREACS~~ ABVS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODE 5 or 6, the ~~ECCS PREACS~~ ABVS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

ACT

~~With the common HEPA filter and charcoal adsorber bank inoperable, the cooling function of the ABVS for ECCS motors is maintained; however, the filtration system function is lost. Since the entire function of the system is not lost, a 24 hour completion time is provided to restore the filters.~~

(continued)

BASES

The 24 hour completion time is acceptable because it is a common filter system and the completion time is shorter than the EGCS completion time. The 24 hour completion time is based on the low probability of a DBA occurring during this time period.

A B.1

With one EGCS PREACS ABVS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the EGCS PREACS ABVS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that for the EGCS (72 hour Completion Time), and this system is not a direct support system for the EGCS. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

Concurrent failure of two EGCS PREACS ABVS trains would result in the loss of functional both filtration and cooling capability; therefore, LCO 3.0.3 must be entered immediately.

BC.1 and BC.2

If the EGCS PREACS ABVS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations dry out any moisture that may have accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated ≥ 10 continuous hours with the heaters energized and operating automatically (filter temperature control). Since the ABVS has only one common charcoal filter, one train needs to be operated for ≥ 10 hours to dry out the filter and the other train only needs to be operated long enough (≥ 15 minutes) to verify all components are operating correctly. Monthly verification of the separate OPERABLE vital power

(continued)

BASES

~~supplies for the exhaust fans, via a non-IS surveillance, assures system redundancy. Systems without heaters need only be operated for > 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.~~

SR 3.7.12.2

This SR verifies that the required ECSS PREACS ABVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTR). The ECSS PREACS ABVS filter tests are in accordance with References 3 and 4. The VFTR includes testing HEPA filter performance, charcoal adsorbers efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTR.

SR 3.7.12.3

This SR verifies that each ECSS PREACS ABVS train starts and operates on an actual or simulated actuation signal and that the system aligns to exhaust through the common HEPA filter and charcoal adsorber. The 18 month Frequency is consistent with that specified in References 3 and 4.

SR 3.7.12.4 Not Used

~~This SR verifies the integrity of the ECSS pump room enclosure. The ability of the ECSS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the ECSS PREACS. During the [post accident] mode of operation, the ECSS PREACS is designed to maintain a slight negative pressure in the ECSS pump room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The ECSS PREACS is designed to maintain a \leq [0.125] inches water gauge relative to atmospheric pressure at a flow rate of [3000] cfm from the ECSS pump room. The Frequency of [18] months is consistent with the guidance provided in NUREG 0800, Section 6.5.1 (Ref. 6).~~

~~This test is conducted with the tests for filter penetration; thus, an [18] month Frequency on a STAGGERED TEST BASIS is consistent with that specified in Reference 4.~~

SR 3.7.12.5 Not Used

~~Operating the ECSS PREACS bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the ECSS PREACS bypass damper is verified if it can be specified in Reference 4.~~

BASES

SURVEILLANCE REQUIREMENT
(continued) SR 3.7.12.6

This SR verifies the leak tightness of dampers that isolate flow to the normally operating filter train. This SR assures that the flow from the auxiliary building passes through the HEPA filter and charcoal absorber unit when the ABVS Buildings and Safeguards or Safeguards Only modes have been actuated coincident with an SI. The 18 month frequency is consistent with the requirements of Reference 4.

REFERENCES

1. FSAR, Section [6.5.1] 9.4.2.
 2. ~~FSAR, Section [9.4.5].~~
 - 3 2. FSAR, Section [15.6.] 15.5.
 - 4 3. ~~Regulatory Guide 1.52 (Rev. 2).~~ ASTM D 3803-1989
 4. ANSI N510-1980
 5. 10 CFR 100.11.
 6. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
 7. DCM S-23B, "Main Auxiliary Building Heating and Ventilation System"
-

B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Handling Building Air Cleanup Ventilation System (FBACS FHBVS)

BASES

BACKGROUND

The FBACS FHBVS filters airborne radioactive particulates and radioactive iodine from the area of the fuel pool following a fuel handling accident or loss of coolant accident (LOCA). The FBACS FHBVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

The FBACS FHBVS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and an exhaust fan. A third non-vital exhaust fan is used for normal operation and has only a prefilter and a HEPA filter. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters, functioning to reduce the relative humidity of the airstream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter fails. The downstream HEPA filter is not credited in the analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal or loss of the normal exhaust fan E-4.

The FBACS FHBVS is a standby system, parts of which may also be operated during normal plant operations. Upon receipt of the actuating signal, normal air discharges from the building, the fuel handling building is isolated and the normal exhaust fan shuts down and the vital exhaust fans start and the stream of ventilation air discharges through the system filter trains. The prefilters or demisters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The FBACS FHBVS is discussed in the FSAR, Sections [6.5.1] 9.4.4, [9.4.5], and [15.7.4] 15.5 (Refs. 1, and 2, and 3, respectively) because it may be used for normal, as well as post (fuel handling) accident, atmospheric cleanup functions.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The FBACS FHBVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3 2, assumes that all fuel rods in an assembly are damaged. The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the FBACS. The DBA analysis of the fuel handling accident assumes that only one train of the FBACS FHBVS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident and for a LOCA. In accordance with assumptions made in the fuel handling accident analysis, loss of offsite power is not considered concurrent with a fuel handling accident. However, loss of power is enveloped by the fuel handling accident analysis. To maximize FHBVS capability to mitigate the consequences of a fuel handling accident, at least one of the FHBVS trains must be capable of being supplied from an operable emergency diesel generator at all times whenever fuel movement is taking place in the spent fuel pool. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4 3).

The FBACS FHBVS satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

Two independent and redundant trains of the FBACS FHBVS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. This requires that when two trains of the FHBVS are OPERABLE, at least one train of the FHBVS must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which energizes the FHBVS train. When only one train is OPERABLE, an OPERABLE diesel generator must be directly associated with the bus which energizes that one OPERABLE FHBVS train. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR 100 (Ref. 5 4) limits in the event of a fuel handling accident.

The FBACS FHBVS is considered OPERABLE when the individual components necessary to control exposure in the releases from fuel handling building are OPERABLE in both trains. An FBACS FHBVS train is considered OPERABLE when its associated:

- a. Exhaust Fan Fan is OPERABLE;

(continued)

BASES

- b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and
 - c. Heater, demister, Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.
-

APPLICABILITY

In MODE 1, 2, 3, or 4, the FBACS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a LOCA and leakage from containment and annulus.

In MODE 1, 2, 3, 4, 5 or 6, the FBACS FHBVS is not required to be OPERABLE since the ECCS is not required to be OPERABLE if provides no safety function associated with these MODES of operation.

During movement of irradiated fuel in the fuel handling areabuilding, the FBACS FHBVS is required to be OPERABLE to alleviate the consequences of a fuel handling accident.

ACTIONS

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1, A.2 and A.3

With one FBACS FHBVS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this period, immediately or place the remaining OPERABLE train is adequate to perform the FBACS function. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable FBACS train, and the remaining FBACS train providing the required protection, in operation and verify that it has an OPERABLE emergency power source, or suspend movement of irradiated fuel assemblies in the fuel handling building. This The suspension of movement of fuel assemblies does not preclude movement of assemblies to a safe position.

(continued)

BASES

B.1 and B.2

~~In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the associated Completion Time, or when both FBACS trains are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.~~

C.1 and C.2

~~When Required Action A.1 cannot be completed within the required Completion Time, during movement of irradiated fuel assemblies in the fuel building, the OPERABLE FBACS train must be started immediately or fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failure will be readily detected.~~

~~If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel handling accident. This does not preclude the movement of fuel assemblies to a safe position.~~

(continued)

BASES

ACTIONS
(continued)

B.1

When two trains of the FBACS FHBVS are inoperable during movement of irradiated fuel assemblies in the fuel handling building, ~~action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel handling building.~~ This does not preclude the movement of fuel assemblies to a safe position.

SURVEILLANCE
REQUIREMENTS

Once actuated due to a fuel handling accident the FHBVS must be protected against a single failure coincident with a loss of offsite power. Protection against a loss of power, although not required for immediate accident response, is assured by requiring that a backup power supply be provided as described above in the LCO section. This back up is assured via the performance of non-IS surveillances.

SR 3.7.13.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

~~Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for > 10 continuous hours with the heaters energized. Systems without heaters need only be operated for > 15 minutes to demonstrate the function of the system.]~~ The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.13.2

This SR verifies that the required FBACS FHBVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The FBACS FHBVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5 and 6). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test-frequencies and additional information are discussed in detail in the VFTP.

(continued)

BASES

SR 3.7.13.3

This SR verifies that each FBACS FHBVS train starts and operates on an actual or simulated actuation signal and directs its exhaust flow through the HEPA Filters and charcoal absorber banks. The 18 month Frequency is consistent with Reference 6.

SR 3.7.13.4

This SR verifies the integrity of the fuel handling building enclosure. The ability of the fuel handling building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FBACS FHBVS. During the [post accident]-mode of operation, the FBACS FHBVS is designed to maintain a slight negative pressure in the fuel handling building, to prevent unfiltered LEAKAGE. The FBACS FHBVS is designed to maintain a the building pressure \leq 0.125 inches water gauge with respect to atmospheric pressure at a flow rate of [20,000]cfm to the fuel building. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 7).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 6.

SR 3.7.13.5 ~~Not Used~~

~~Operating the FBACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FBACS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with Reference 6.~~

(continued)

BASES

REFERENCES

1. ~~FSAR, Section [6.5.1].~~
- 2 ~~1.~~ FSAR, Section [9.4.5 ~~4~~].
- 3 ~~2.~~ FSAR, Section [~~15.7.4~~] ~~15~~5.
- 4 ~~3.~~ Regulatory Guide 1.25.
- 5 ~~4.~~ 10 CFR 100.
- 6 ~~5.~~ Regulatory Guide ~~1.52 (Rev. 2).~~ ~~ASTM D 3802-1989~~
6. ~~ANSI N510-1980~~
7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
8. ~~DCM S-23D, Fuel handling Building HVAC System~~

B 3.7 PLANT SYSTEMS

B 3.7.14 ~~Not Used~~ Penetration Room Exhaust Air Cleanup System (PREACS)BASES

BACKGROUND

~~The PREACS filters air from the penetration area between containment and the auxiliary building.~~

~~The PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation, as well as demisters, functioning to reduce the relative humidity of the air stream, also form part of the system. A second bank of HEPA filters, which follows the adsorber section, collects carbon fines and provides backup in case of failure of the main HEPA filter bank. The downstream HEPA filter, although not credited in the accident analysis, collects charcoal fines and serves as a backup should the upstream HEPA filter develop a leak. The system initiates filtered ventilation following receipt of a safety injection signal.~~

~~The PREACS is a standby system, parts of which may also operate during normal unit operations. During emergency operations, the PREACS dampers are realigned and fans are started to initiate filtration. Upon receipt of the actuating signal(s), normal air discharges from the penetration room, the penetration room is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, as well as any entrained water droplets, to prevent excessive loading of the HEPA filters and charcoal adsorbers.~~

~~The PREACS is discussed in the FSAR, Sections [6.5.1], [9.4.5], and [15.6.5] (Refs. 1, 2, and 3, respectively) since it may be used for normal, as well as post accident, atmospheric cleanup functions. Heaters may be included for moisture removal on systems operating in high humidity conditions. The primary purpose of the heaters is to maintain the relative humidity at an acceptable level consistent with iodine removal efficiencies per Regulatory Guide 1.52 (Ref. 4).~~

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

~~The PREACS design basis is established by the large break loss of coolant accident (LOCA). The system evaluation assumes a passive failure outside containment, such as valve packing leakage during a Design Basis Accident (DBA). In such a case, the system restricts the radioactive release to within the 10 CFR 100 (Ref. 4) limits, or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The analysis of the effects and consequences of a large break LOCA are presented in Reference 3.~~

~~Two types of system failures are considered in the accident analysis: a complete loss of function, and excessive LEAKAGE. Either type of failure may result in less efficient removal of any gaseous or particulate material released to the penetration room following a LOCA.~~

~~The PREACS satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).~~

LCO

~~Two independent and redundant trains of the PREACS are required to be OPERABLE to ensure that at least one train is available, assuming there is a single failure disabling the other train coincident with a loss of offsite power.~~

~~The PREACS is considered OPERABLE when the individual components necessary to control radioactive releases are OPERABLE in both trains. A PREACS train is considered OPERABLE when its associated:~~

- ~~a. Fan is OPERABLE;~~
- ~~b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and~~
- ~~c. Heater, demister, ductwork, valves, and dampers are OPERABLE and air circulation can be maintained.~~

APPLICABILITY

~~In MODES 1, 2, 3, and 4, the PREACS is required to be OPERABLE, consistent with the OPERABILITY requirements of the Emergency Core Cooling System (ECCS).~~

~~In MODE 5 or 6, the PREACS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.~~

ACTIONS

A-1

(continued)

BASES

~~With one PREACS train inoperable, the action must be taken to restore OPERABLE status within 7 days. During this period, the remaining OPERABLE train is adequate to perform the PREACS function. The 7 day Completion Time is appropriate because the risk contribution of the PREACS is less than that of the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7-day Completion Time is based on the low probability of a DBA occurring during this period, and the remaining train providing the required capability.~~

B.1 and B.2

~~If the inoperable train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.~~

SURVEILLANCE
REQUIREMENTS

SR 3.7.14.1

~~Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation dries out any moisture that may have accumulated in the charcoal as a result of humidity in the ambient air. [Systems with heaters must be operated for > 10 continuous hours with the heaters energized. Systems without heaters need only be operated for > 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known~~

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

~~SR 3.7.14.1 (continued)~~

~~reliability of equipment and the two train redundancy available.~~

~~SR 3.7.14.2~~

~~This SR verifies that the required PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The PREACS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5). The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].~~

~~SR 3.7.14.3~~

~~This SR verifies that each PREACS starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 5.~~

~~SR 3.7.14.4~~

~~This SR verifies the integrity of the penetration room enclosure. The ability of the penetration room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of PREACS. During the [post accident] mode of operation, the PREACS is designed to maintain a \leq [0.125] inches water gauge relative to atmospheric pressure at a flow rate of [3000] cfm in the penetration room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The Frequency of [18] months is consistent with the guidance provided in NUREG 0800 (Ref. 6).~~

~~The minimum system flow rate maintains a slight negative pressure in the penetration room area, and provides sufficient air velocity to transport particulate~~

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

~~SR 3.7.14.4 (continued)~~

~~contaminants, assuming only one filter train is operating. The number of filter elements is selected to limit the flow rate through any individual element to about [3000] cfm. This may vary based on filter housing geometry. The maximum limit ensures that the flow through, and pressure drop across, each filter element are not excessive.~~

~~The number and depth of the adsorber elements ensure that, at the maximum flow rate, the residence time of the air stream in the charcoal bed achieves the desired adsorption rate. At least a [0.125] second residence time is necessary for an assumed [99]% efficiency.~~

~~The filters have a certain pressure drop at the design flow rate when clean. The magnitude of the pressure drop indicates acceptable performance, and is based on manufacturers' recommendations for the filter and adsorber elements at the design flow rate. An increase in pressure drop or a decrease in flow indicates that the filter is being loaded or that there are other problems with the system.~~

~~This test is conducted along with the tests for filter penetration; thus, the [18] month Frequency is consistent with that specified in Reference 5.~~

~~SR 3.7.14.5~~

~~It is necessary to operate the PREACS filter bypass damper to ensure that the system functions properly. The OPERABILITY of the PREACS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with that specified in Reference 5.~~

REFERENCES

- ~~1. FSAR, Section [6.5.1].~~
- ~~2. FSAR, Section [9.4.5].~~
- ~~3. FSAR, Section [15.6.5].~~
- ~~4. 10 CFR 100.~~
- ~~5. Regulatory Guide 1.52, Rev. 2.~~
- ~~6. NUREG 0800, Section 6.5.1, Rev. 2, July 1981.~~

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel storage pool design is given in the FSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section [15.7.4], 15.4.5 and 15.5.22 (Ref. 3).

APPLICABLE
SAFETY ANALYSES

The minimum water level in the spent fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle rods and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. Although there are other spent fuel pool elevations where fuel handling accidents can occur, the design basis fuel handling accident, which uses the conservative assumptions of RG 1.25, is expected to be bounding. To add conservatism, the analysis assumes that all fuel rods of the damaged fuel assembly fail. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

In practice, the water level maintained for fuel handling provides more than 23 feet of water over the top of irradiated fuel assemblies seated in the storage racks. FSAR Section 9.1.4.3.4 requires the water level to be at or above 137 feet 8 inches during fuel handling to assure 8 feet of water shielding.

(continued)

BASES

This water level corresponds to 24 feet 6 inches above the top of the fuel assemblies in the racks and to 23 feet above a fuel assembly lying horizontally on top of the racks.

The spent fuel storage pool water level satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

The spent fuel storage pool water level is required to be \geq 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel storage pool, since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel storage pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE
REQUIREMENTS

SR 3.7.15.1

This SR is done during the movement of irradiated fuel assemblies as stated in the Applicability. This SR verifies sufficient fuel

(continued)

BASES

storage pool water is available in the event of a fuel handling accident. The water level in the spent fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the spent fuel storage pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with ~~SR 3.9.6.1~~ SR 3.9.7.1.

REFERENCES

1. FSAR, Section 9.1.2.
 2. FSAR, Section 9.1.3.
 3. FSAR, Section ~~[15.7.4]~~ 9.1.4.3.4, 15.4.5 and 15.5.22.
 4. Regulatory Guide 1.25, Rev 0.
 5. 10 CFR 100.11.
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B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

In the Maximum Density Rack (MDR) [(Refs. 1 and 2)] design, The spent fuel storage pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. [Region 1], with [336] 290 storage positions, is designed to accommodate has been analyzed for the storage of new fuel with a maximum enrichment of [4.65] wt% U-235, or spent fuel regardless of the discharge fuel burnup assemblies which meet the requirements of LCO 3.7.17.1. [Region 2], with [2670] 1034 storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure [3.7.17.12], in the accompanying has been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17.2. Fuel assemblies not meeting the criteria of Figure [3.7.17] shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify require that the limiting k_{eff} is at or below the limit of 0.95 be evaluated in the absence of soluble boron. Hence, the design analysis of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 31) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenario is associated with the movement of fuel from [Region 1 to Region 2], and accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the to criticality of [Region 2]. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.17, "Spent Fuel Assembly Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.

BASES

APPLICABLE
SAFETY ANALYSES

Most accident conditions do not result in an increase in the negligible reactivity effect for either of the two regions (Ref. 2 and 3). Examples of these accident conditions are the loss of cooling (reactivity increase with decreasing water density) and the dropping of a fuel assembly on the top of the rack. However, accidents scenarios can be postulated that could have more than a negligible positive reactivity effect increase the reactivity. One such scenario is associated with placing a fuel assembly which is required to be stored in Region 1 in Region 2. This could potentially increase the k_{eff} of Region 2 above 0.95 reactivity is unacceptable with unborated water in the storage pool. Thus, to compensate for reductions in the subcriticality margin from postulated these accident occurrences conditions, the presence of spent fuel pool contains 2000 ppm soluble boron in the pool prevents criticality in both regions. The postulated accidents are basically of two types. A fuel assembly could be incorrectly transferred from [Region 1 to Region 2] (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). The second type of postulated accidents is associated with a fuel assembly which is dropped adjacent to the fully loaded [Region 2] storage rack. This could have a small positive reactivity effect on [Region 2]. However, The negative reactivity effect of the soluble boron more than compensates for the increased reactivity caused by either one of the two postulated accident scenarios. The accident analyses is provided in the FSAR, Section [15.7.4] 15.5.22 (Ref. 4).

The concentration of dissolved boron in the spent fuel storage pool satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

The spent fuel storage pool boron concentration is required to be \geq [2300] 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in Reference 4 2, 3, and 4. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel storage pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool, until a complete spent fuel storage pool verification has been performed following the last movement of

(continued)

BASES

~~fuel assemblies in the spent fuel storage pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.~~

ACTIONS

A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the ~~spent~~ fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by ~~immediately suspending the movement of fuel assemblies and immediately taking actions to restore the spent fuel pool boron concentration to greater than or equal to 2000 ppm. The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. An acceptable alternative is to verify by administrative means that the fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. This suspension of fuel movement does not preclude movement of a fuel assemblies to a safe position.~~

If the LCO is not met while moving irradiated fuel assemblies in ~~MODE 5 or 6~~, LCO 3.0.3 would not be applicable. ~~If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, since the inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE
REQUIREMENTS

SR 3.7.16.1

This SR verifies ~~by chemical analysis~~ that the concentration of boron in the ~~spent~~ fuel storage pool is ~~within at or above~~ the required limit. AS long as this SR is met, the analyzed accidents are fully addressed. The ~~7 31~~ day Frequency is appropriate because no major replenishment of pool water is expected to take place ~~ever such a short period of time.~~

(continued)

BASES

REFERENCES

1. ~~Callaway FSAR, Appendix 9.1A, "The Maximum Density Rack (MDR) Design Concept."~~
 2. ~~Description and Evaluation for Proposed Changes to Facility Operating Licenses DPR 39 and DPR 48 (Zion Power Station).~~
 3. ~~1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).~~
 4. ~~2. FSAR, Section [15.7.4] "Criticality Safety Evaluation of Region 1 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment." S.E. Turner, October 1993, Holtec Report HI-931076.~~
 3. ~~"Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment." S.E. Turner, October 1993, Holtec Report HI-931077.~~
 4. ~~FSAR, Section 9.1 and 15.2.22.~~
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B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

~~In the Maximum Density Rack (MDR) [(Refs. 1 and 2)] design, The spent fuel storage pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. [Region 1], with [336] 290 storage positions, is designed to accommodate [has been analyzed for the storage of new fuel with a maximum enrichment of [4.65] wt% U-235, or spent fuel regardless of the discharge fuel burnup assemblies which meet the requirements of LCO 3.7.17.1. [Region 2], with [2670] 1034 storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure [3.7.17.12], in the accompanying [has been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17.2. Fuel assemblies not meeting the criteria of Figure [3.7.17.1] shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.~~

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, ~~specify require~~ that the limiting k_{eff} of the fuel configuration is at or below the limit of 0.95 be evaluated in the absence of soluble boron. Hence, the design analysis of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 3) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. ~~For example, the most severe accident scenario is associated with the movement of fuel from [Region 1 to Region 2], and accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the criticality of [Region 2]. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with the accompanying LCO.~~

Prior to movement of an assembly, it is necessary to ~~verify that perform SR 3.7.16.1 is current.~~

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The hypothetical accidents can only take place during or as a result of the movement of an assembly. The analyzed accidents that could have significant reactivity effects are the misplacement of a fuel assembly, a significant increase in spent fuel pool temperature above the design basis temperature of 150°F, or a cask drop accident (Ref. 4 2, 3, and 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.16, "Spent Fuel Storage Pool Boron Concentration") ensures that k_{eff} will remain at or below 0.95 prevents criticality in both regions. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time. During the remaining time period with no potential for accidents, the operation may be under the auspices of the accompanying LCO.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with LCO 3.7.17.1 and 3.7.17.2 Figure 3.7.17.1 or 3.7.17.2, in the accompanying LCO, ensures the k_{eff} of the spent fuel storage pool will always remain ≤ 0.95 , assuming the pool to be flooded with unborated water at a temperature of $\leq 150^\circ\text{F}$. Fuel assemblies not meeting the criteria of Figure [3.7.17.1] shall be stored in accordance with Specification 4.3.1.1 in Section 4.3.

APPLICABILITY

This These LCOs applies apply whenever any fuel assembly is stored in [Region 2] of the spent fuel storage pool.

ACTIONS

A.1

The Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since the inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in [Region 2] the spent fuel storage pool is not in accordance with Figure 3.7.17.1 LCO 3.7.17.1 and 3.7.17.2, or paragraph 4.3.1.1, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 3.7.17.1 LCO 3.7.17.1 and 3.7.17.2 or

(continued)

BASES

~~Specification 4.3.1.1 which will return the fuel pool to an analyzed condition.~~

~~If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.~~

SURVEILLANCE
REQUIREMENTS

~~SR 3.7.17.1.1 and SR 3.7.17.1.2~~

~~These SRs verify by administrative means that the each fuel assembly and its expected storage location are in accordance with the applicable LCO (3.7.17.1 or 3.7.17.2), prior to each fuel assembly move when the assembly is to be stored in Region 1 or 2 of the spent fuel pool. Initial enrichment and burnup of the fuel assembly is in accordance with Figure [3.7.17.1] in the accompanying LCO. For fuel assemblies in the unacceptable range of Figure 3.7.17.1, performance of this SR will ensure compliance with Specification 4.3.1.1.~~

BASES

REFERENCES

1. ~~Callaway FSAR, Appendix 9.1A, "The Maximum Density Rack (MDR) Design Concept."~~
 2. ~~Description and Evaluation for Proposed Changes to Facility Operating Licenses DPR 39 and DPR 48 (Zion Power Station).~~
 3. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
 4. FSAR, Section ~~[15.7.4]~~ 15.5.22.
 3. "Criticality Safety Evaluation of Region 1 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment." S. E. Turner, October 1993. Holtec Report HI-931076.
 4. "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment." S. E. Turner, October 1993. Holtec Report HI-931076.
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B 3.7 PLANT SYSTEMS

B 3.7.18 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 $\mu\text{Ci/gm}$ (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses. ~~Operating at or below 0.1 $\mu\text{Ci/gm}$ ensures that in the event of a DBA, offsite doses will be less than 10 CFR 100 requirements.~~

~~With the specified activity limit, the resultant 2-hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.58 rem if the main steam safety valves (MSSVs) open for 2 hours following a trip from full power.~~

~~Operating a unit at the allowable limits could result in a 2-hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.~~

APPLICABLE
SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a

(continued)

BASES

~~small fraction of the unit EAB10 CFR 100~~ limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADV). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generators ~~are~~ ~~is~~ assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. ~~Since no credit is taken in the analysis for activity plateout or retention, the quantity of radioactivity released to the environment due to an SGTR depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow flashing fractions, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the steam generator, and liquid-vapor partitioning in the condenser hotwell. All of these parameters were conservatively evaluated in a manner consistent with the recommendations of Standard Review Plan Section 15.6.3 and the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.~~

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be $\leq 0.10 \mu\text{Ci/gm DOSE EQUIVALENT I-131}$ to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner

BASES

to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.18.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.
 2. FSAR, Chapter 15.
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Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(5 Pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG 1431

NUREG-1431 Section 3.7

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431 to make them plant specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER

JUSTIFICATION

3.7-01

ACTION A.1 is revised, new ACTION A.2 is added, and Table 3.7.1-1 is revised consistent with Traveler WOG-83, Rev. 0 to account for the fact that a reduction in power level is not directly proportional to the reduction in main steam safety valves (MSSV) relieving capability and plants which may operate for some part of a fuel cycle with a positive moderator temperature coefficient (MTC). Per Westinghouse Nuclear Safety Advisory Letter, NSAL 94-001, if the MTC is positive at the required reduced power level, the reactor coolant system (RCS) heat up following a turbine trip event could result in a core power increase and additional heat transfer to the secondary system which may not be attenuated without over pressurizing the main steam system. To preclude this condition the power range neutron flux high trip set point is required to be reset to a power level consistent with the number of inoperable safety valves within 72 hours. [A Note is added that states that Required Action A.2 is only applicable in MODE 1]. These changes are consistent with Westinghouse Owners Group (WOG) Traveler WOG-83 and NSAL 94-001.

A recent revision to WOG-83 (Rev.1) has been proposed requiring that the power range neutron flux trip high setpoints be reduced when at a reduced reactor power level to account for a control rod withdrawal event at reduced reactor power. The identification of this issue has identified a non-conservatism in NUREG-1431. Consequently, the requirement in the CTS to reduce the power range neutron flux trip high setpoints with inoperable MSSVs [regardless of the value of MTC] is retained. However, the 72 hour Completion Time proposed in the traveler is incorporated into the ITS. These changes are acceptable because the retention of the requirement to reduce the power range flux trip high setpoint is more conservative than NUREG-1431 or WOG-83 and the extended Completion Time recognizes the low probability of an event occurring during the 72 hours allowed to reset the trip setpoints.

3.7-02

Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B)

3.7-03

SR 3.7.3.1 is divided into two surveillances since both the stroke time and the surveillance Frequency requirements are different for the feedwater regulation and associated bypass valves and the feedwater isolation valves.

3.7-04

Not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B)

**CHANGE
NUMBER**

JUSTIFICATION

- 3.7-05 Required Action B.1 is revised to state and new C.1 states respectively that restoration of "all but one" and "all but two" atmospheric [dump] valves [(ADV)] is required which will effectively exit the respective Required Actions. This is consistent with the Bases of NUREG-1431. The specific change proposed by Industry Traveler TSTF-100 is to add the "all but one" phrase to Required Action B for plants that only require three [ADV]s to be OPERABLE. The addition of the "all but two" phrase to new Required Action C.1 is to account for the requirement to have four [ADV]s OPERABLE.
- 3.7-06 The Condition and Required Action for two or more inoperable [ADV] lines is revised to limit the applicability to only two inoperable [ADV] lines and the Completion Time is revised from 24 to 72 hours per the current licensing basis. A new Condition C for three or more [ADV] lines inoperable for plants that require four [ADV] lines is proposed. The original Condition C is retained and relabeled as Condition D. These changes are consistent with the intent of NUREG 1431.
- 3.7-07 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-08 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-09 New Conditions, F, G, and H, and the SR associated with the fire water storage tank (FWST) AFW pump supply are relocated from the CTS on AFW supply and included in the AFW specification for completeness.
- 3.7-10 Consistent with the CTS, ITS 3.7.6 specification description, the LCO, the ACTION requirements and the Surveillances are revised to incorporate the specific requirement of an OPERABLE AFW supply source via the CST and the FWST. The volumes in gallons are converted to the equivalent level in tank percent full. The control room readouts are in percent level.
- 3.7-11 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B)
- 3.7-12 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B)
- 3.7-13 ITS 3.7.8, ACTION A.1, Note 1, under Required Action A.1 is deleted consistent with the CTS. The emergency diesel generators have no interface with the auxiliary saltwater (ASW) system or the closed cooling water (CCW) system; each diesel has its own self contained cooling system.
- 3.7-14 Consistent with the CTS, the ITS SR 3.7.8.1 Note which states that isolation of individual components does not render the [ASW] inoperable is deleted since the [ASW] system supplies only the CCW heat exchanger and no other individual components.

**CHANGE
NUMBER**

JUSTIFICATION

- 3.7-15 SR 3.7.8.1 is revised to include a requirement to verify that power and air is available such that the valves may be placed in their required positions as described in the Bases. The normal operating configuration may require reconfiguration for an ASW pump failure or for train separation for long term heat removal following a DBA. The OPERABILITY defined in the LCO states valves must be capable of performing their safety-related function. Motive force is required for the valves to be OPERABLE.
- 3.7-16 SR 3.7.8.2 is revised to include only power operated valves since there are no automatically operated valves in the ASW system. The surveillance is revised to require movement of power operated valves to demonstrate the ability to reconfigure the ASW system as described in the FSAR and the ITS Bases. These changes assure continued compliance with the existing licensing basis.
- 3.7-17 The UHS specification is revised, per the current licensing basis, to specify that for the system to perform its intended function, it is temperature limited. The Required Action A is revised to include the CTS license basis requirements which encompasses ACTION B, thus the ACTION B inoperable statement is deleted. The surveillance requirements are revised per the CTS and renumbered with respect to temperature limits and frequencies and the remaining NUREG-1431 surveillances that are not applicable are deleted.
- 3.7-18 The CTS surveillance for leakage testing of ABVS dampers M2A and M2B is retained.
- 3.7-19 Not applicable to DCCP, See Conversion Comparison Table (Enclosure 6B).
- 3.7-20 Not applicable to DCCP, See Conversion Comparison Table (Enclosure 6B).
- 3.7-21 The ACTIONS are revised to include the CTS action requirement for an inoperable HEPA filter and/or charcoal absorber. SR 3.7.12.1 is revised to reflect the plant design of one common charcoal absorber bank and the appropriate charcoal absorber drying time.
- 3.7-22 SR 3.7.12.3 is revised consistent with the CTS, to describe the expected actions upon an actuation of the ABVS. This revision reflects the plant design of one common HEPA filter and charcoal absorber bank, and the need to verify alignment for flow through the bank.
- 3.7-23 ACTION C and SR 3.7.12.4 are deleted consistent with the CTS, since the ABVS was not designed to maintain a specific negative pressure. The system is designed and balanced to maintain building inflow, but not at a specific negative pressure.
- 3.7-24 SR 3.7.12.5 is deleted consistent with the CTS, since there are no ABVS bypass dampers and the dampers that activate to align the system to the common HEPA filter charcoal absorber bank are tested by SR 3.7.12.3.
- 3.7-25 Not applicable to DCCP. See Conversion Comparison Table (Enclosure 6B).

**CHANGE
NUMBER**

JUSTIFICATION

- 3.7-26 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-27 A Note is added to Table 3.7.1-2 under LIFT SETTING that specifies that the lift point of the lowest set safety is +3% and -2%.
- 3.7-28 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-29 Revise AFW pump testing Frequency to be "In accordance with Inservice Test Program." These changes are consistent with TSTF-101, and will eliminate any ambiguity associated with pump testing as a result of ASME changes.
- 3.7-30 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-31 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-32 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-33 The requirement to verify a make-up flow rate during the tests demonstrating the capability to maintain control room differential pressure above atmospheric pressure would be deleted. The current licensing basis of the plant is to be able to maintain a positive pressure in the control room with respect to the outside atmosphere.
- 3.7-34 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-35 SR 3.7.10.3 is revised to reflect plant configuration and current licensing basis required testing.
- 3.7-36 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-37 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-38 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-39 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-40 Not used.
- 3.7-41 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-42 This change adds a Note that states that LCO 3.0.3 is not applicable to the fuel handling building ventilation system during fuel movement since fuel movement is independent of reactor operation. This exemption is part of the DCPP CTS and has been proposed as a generic change to NUREG-1431 by Industry Traveler TSTF-36, Rev. 2.
- 3.7-43 ACTION A is revised and ACTIONS C and E are not used per the current licensing basis. The FHBVS for the plant does not act as part of the ventilation system used to filter post LOCA leakage external to the containment.

**CHANGE
NUMBER**

JUSTIFICATION

- 3.7-44 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-45 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-46 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-47 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-48 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.7-49 The requirement to verify a make-up flow rate during the tests demonstrating the capability to maintain [fuel handling] building differential pressure below atmospheric pressure would be deleted. The current licensing basis of the plant is to be able to maintain a negative pressure [in the fuel handling building] with respect to the outside atmosphere.
- 3.7-50 The CTS ADV surveillance that verifies the back-up air bottle pressure once per 24 hours is retained.
- 3.7-51 A new spent fuel pool storage specification is created for Region 1 fuel storage due to the unique storage requirements.
- 3.7-52 ITS 3.7.11 is not used due to the mild coastal environment in which DCPP is located, consistent with the CTS.
- 3.7-53 ITS 3.7.16 is revised to be consistent with the current licensing basis and CTS. The boron concentration is required to be within limits whenever fuel is stored in the spent fuel pool to prevent an increase in the k_{eff} of the racks above 0.95 should the spent fuel pool temperature increase above 150°F. The Frequency for verification of the boron concentration is changed from 7 days to 31 days consistent with the CTS.
- 3.7-54 The LCO, Required Actions, and Surveillances are revised per the CTS. The CTS evaluates Region 2 fuel storage on fuel pellet diameter and a checker board loading pattern in addition to the other NUREG-1431 requirements.
- 3.7-55 The NUREG-1431 3.7.1.4 specification is not used since an equivalent safety grade system does not exist. Therefore, the deletion is per the current licensing basis.
- 3.7-56 This change creates a new SR for the MSIVs and MFIVs to distinguish between the IST and the automatic actuation testing of these isolation valves. The surveillance allows credit for an actual actuation, if one occurs, to satisfy the surveillance requirements. These changes are consistent with WOG-98. Although SRs 3.7.2.2 and 3.7.3.2 are new SRs, they may be performed in conjunction with SRs 3.7.2.1 and 3.7.3.1. Therefore, the Note allowing testing to be performed on MODE 3 is also needed for these new SRs.
- 3.7-57 Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(8 Pages)

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-01	ACTION A.1 is revised [new ACTION A.2 is added , and Table 3.7.1-1 is revised] to account for operation with inoperable MSSVs and resetting the power range neutron flux high trip setpoints with inopreable MSSVs. The Compeltion Time to reset the power range neutron flux high trip setpoints is extended to 72 hours. [The table is revised based on NSAL 94-001 and deletes the reference to five OPERABLE MSSVs.]	Yes	Yes	Yes	YES
3.7-02	The CTS Applicability of MODES 1, 2, and 3 is being retained in ITS 3.7.2, MSIVs, and ITS 3.7.3, MFIVs.	No, DCPD is adopting ITS.	No, CPSES is adopting ITS.	Yes	YES
3.7-03	SR 3.7.3.1 is divided into two surveillances since both the stroke time and the Frequency requirements are different at DCPD for the feedwater regulation/bypass valves and the feedwater isolation valve.	Yes, per LA 7776	No	No	No
3.7-04	Requirements involving reliance on the SG heat removal system for heat removal in MODE 4 would be deleted.	No, DCPD is adopting ITS.	Yes	Yes	Yes
3.7-05	Required Action B.1 and new C.1 are revised to state that restoration of "all but one" and "all but two" [ADV] lines is required which will effectively exit the respective Required Action.	Yes	Yes	No, refer to 3.7-19.	No, refer to 3.7-19.
3.7-06	The Condition and Required Action for two or more inoperable [ADV] lines is limited to two [ADV] lines and the Completion Time is revised from 24 to 72 hours per the current licensing basis. A new Condition C is added.	Yes	Yes	No, not part of CTS.	No, not part of CTS.
3.7-07	Revised Conditions A and C to be consistent with CTS. The ITS as written would have allowed the OPERABLE EES train to remain in standby during movement of irradiated fuel.	No, not part of CTS.	No, not part of CTS.	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-08	SR 3.7.5.1 is revised to add a Note consistent with the CTS and the plant specific design. The verification of flow control valve position is deferred until conditions are appropriate.	No, AFW valves have a correct position.	Yes	Yes	Yes
3.7-09	New Conditions F, G, and H and the surveillance requirement associated with the FWST AFW pump supply are relocated from the CTS on AFW supply and included in the DCPD AFW specification for completeness.	Yes	No	No	No
3.7-10	The specification description, the LCO, the ACTION requirements and the Surveillance are revised to incorporate the DCPD plant specific requirement for OPERABLE AFW supply sources via the CST and the FWST per the current licensing basis.	Yes	No	No	No
3.7-11	The Required Actions for CPSES feedwater isolation and associated bypass valves inoperable are revised consistent with the current licensing basis for a Completion Time of 4 hours and to credit the MFRVs (feedwater control valves (FCVs)) and associated bypass valves for a Completion Time of 72 hours. A new SR is added for the FCVs and associated bypass valves.	No	Yes	No	No
3.7-12	WOG-83 revised Condition A and Table 3.7-1 to account for plants that credit the Power Range High Neutron Flux trip function when MTC is positive (See 3.7-01 above). The wording of the traveler has been modified for CPSES to account for plant specific differences.	No	Yes	No	No
3.7-13	Note 1. under Action Required A.1 is deleted. The DCPD emergency diesel generators have self contained cooling systems that do not rely upon an external source of cooling water.	Yes	No	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-14	The Note for SR 3.7.8.1 is deleted since the DCPD ASW system supplies only the CCW heat exchanger and no other individual components.	Yes	No	No	No
3.7-15	SR 3.7.8.1 is revised to include a DCPD specific requirement to verify the availability of power and air so that the valves can be placed in their correct positions as described in the Bases.	Yes	No	No	No
3.7-16	SR 3.7.8.2 is revised to include only power operated valves since there are no automatically operated valves in the DCPD ASW system. The surveillance is revised to require movement of power operated valves to demonstrate the ability to reconfigure the ASW system as described in the FSAR and the ITS Bases.	Yes	No	No	No
3.7-17	The DCPD UHS specification is revised to reflect the fact that for the system to perform its intended function, it is temperature limited. The LCO is revised to note that the specification limits the temperature to less than or equal to 64°F. SR numbers are revised accordingly and the remaining surveillances are deleted as being not applicable.	Yes	No	No	No
3.7-18	The DCPD specific CTS surveillance for leakage testing of ABVS dampers M2A and M2B is retained.	Yes	No	No	No
3.7-19	Required Action B.1 is revised to state that restoration of "all but" one [ASD] line is required, which will effectively exit Required Action B.1 and re-enter Required Action A.1.	No, refer to change 3.7-5 and 3.7-6.	No, refer to change 3.7-5 and 3.7-6.	Yes	Yes
3.7-20	A Callaway specific Condition is added to address the inoperability of one of the Essential Service Water (ESW) supplies to the turbine-driven AFW pump.	No	No	No	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-21	The ACTIONS and SR 3.7.12.1 are revised to include the DCPD specific CTS ACTION requirement for an inoperable HEPA filter and/or charcoal absorber and to provide the appropriate charcoal absorber monthly drying time for the single common charcoal absorber.	Yes	No	No	No
3.7-22	SR 3.7.12.3 is revised for DCPD to describe the expected actions upon an actuation of the ABVS.	Yes	No	No	No
3.7-23	ACTION C and SR 3.7.12.4 are deleted since the DCPD ABVS was not designed to maintain a specific negative pressure.	Yes	No	No	No
3.7-24	SR 3.7.12.5 is deleted since there are no DCPD ABVS bypass dampers and the system automatic dampers are tested by SR 3.7.12.3.	Yes	No	No	No
3.7-25	Based on the CTS, a Note is added to [SR 3.7.3.1, 3.7.3.2, 3.7.4.1 and 3.7.4.2] to indicate that demonstration of valve OPERABILITY is only required to be performed for entry into (and continued operation in) MODES 1 and 2. This Note states that the SR is only required to be performed in MODES 1 and 2. This would allow entry into MODE 3 for the purpose of testing the valves.	No, note is not part of CTS.	No, note is not part of CTS.	Yes	Yes
3.7-26	Condition D is deleted to reflect the CPSES plant specific design of primary FIVs and associated bypass valves and isolation backup via the in series FCVs and associated bypass valves.	No	Yes	No	No
3.7-27	A Note is added to DCPD Table 3.7.1-2 under Lift Setting that specifies that the lift point of the lowest set safety is +3% and -2%.	Yes (per LA 108/107)	No	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-28	Revise [ADV] Frequency from 18 months to "in accordance with Inservice Test Program."	No, CTS is 18 months.	Yes	Yes	Yes
3.7-29	Revise AFW pump testing to be "In accordance with Inservice Test Program."	Yes	Yes	Yes	Yes
3.7-30	LCO 3.7.8 and ACTIONs are revised for CPSES to incorporate requirements for two units with station service water system cross connections.	No, covered by ECG per GL 91-13 response.	Yes	No, single unit plant.	No, single unit plant.
3.7-31	SR 3.7.8.2 is replaced with the current CPSES specific surveillance of the cross connections between units. The CPSES design has no automatic valves as per this SR in the ITS.	No, refer to 3.7-15 and 3.7-16.	Yes	No	No
3.7-32	CONDITION A for CPSES is changed to "SSI level less than required" and SR 3.7.9.3 and 3.7.9.4 are deleted.	No	Yes	No	No
3.7-33	The requirement to verify a make-up flow rate during the tests demonstrating the capability to maintain control room differential pressure above atmospheric pressure would be deleted.	Yes, per CTS.	No, retained CTS requirement.	Yes	Yes
3.7-34	In accordance with Traveler WOG-64, the Completion Time for closing one inoperable MSIV is extended to 72 hours; and separate Required Actions are included for either one MSIV inoperable or two or more MSIVs inoperable in MODES 2 and 3.	No, adopting 8 hour AOT from STS.	Yes	Yes	Yes
3.7-35	SR 3.7.10.3 is revised to reflect DCCP specific plant configuration and CTS required testing.	Yes	No	No	No
3.7-36	Required Actions D and E are revised for CPSES for two trains inoperable where at least 100% of the required heat removal capacity is available.	No	Yes	No	No

DCCP Conversion Comparison Table - Improved TS

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-37	Modifies LCO 3.7.2 Condition A and adds new Condition B and C to be consistent with the CPSES CTS.	No	Yes	No	No
3.7-38	This proposed change deletes reference to a specific flowrate for conducting the negative pressure test per the CPSES CTS.	No, see CN 3.7-49.	Yes	No, see CN 3.7-49.	No, see CN 3.7-49.
3.7-39	SR 3.7.12.6 is added to verify the shutdown of the non-ESF fans to prevent bypass of the ESF Filtration units (CPSES specific).	No	Yes	No	No
3.7-40	Not used.	N/A	N/A	N/A	N/A
3.7-41	The Main Feedwater Regulating and associated Bypass Valves are deleted from the ITS per current licensing basis.	No, CTS includes MFRVs.	No, refer to 3.7-11.	Yes	Yes
3.7-42	Add DCPD specific note that states that 3.0.3 is not applicable to the fuel handling building ventilation system during fuel movement since fuel movement is independent of reactor operation.	Yes	No	No	No
3.7-43	ACTION A of ITS 3.7.13 is revised and ACTIONS C, E and F.1 of ITS 3.7.13 are not used per the DCPD CTS.	Yes	No	No	No
3.7-44	This change would revise ITS 3.7.13 to add a new Note to the Applicability and change the Conditions, Required Actions, and SRs to conform to the design of the Emergency Exhaust System.	No, fuel building ventilation not required for post LOCA leakage.	No, CTS does not require this specification.	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-45	ITS 3.7.15 is revised to be CPSES specific to address the two spent fuel pools and the in-containment storage racks.	No	Yes	No	No
3.7-46	Revised to delete "irradiated fuel assemblies seated in" since accident analysis assumes fuel assembly lying on top of the fuel storage racks.	No, ITS is consistent with CTS.	Yes	No, ITS is consistent with CTS.	Yes
3.7-47	This change adds TS 3.7.19, a safety chilled water system which is in the CPSES CTS.	No	Yes	No	No
3.7-48	This change adds TS 3.7.20, an UPS HVAC system which is in the CPSES CTS.	No	Yes	No	No
3.7-49	The requirement to verify a make-up flow rate during the tests demonstrating the capability to maintain [fuel handling] building differential pressure below atmospheric pressure would be deleted. The current licensing basis of the plant is to be able to maintain a negative pressure [in the fuel handling building] with respect to the outside atmosphere.	Yes	No, see CN 3.7-38.	Yes	Yes
3.7-50	The CTS DCPD specific ADV surveillance that verifies the back-up air bottle pressure once per 24 hours is retained.	Yes	No	No	No
3.7-51	A new spent fuel pool storage specification is created for Region 1 fuel storage due to unique storage requirements at DCPD.	Yes	No	No	No
3.7-52	ITS 3.7.14 is not used due to the mild coastal environment in which the plant is located.	Yes	No	No	No
3.7-53	ITS 3.7.16 for DCPD is revised to be consistent with the current licensing basis and CTS.	Yes	No	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.7

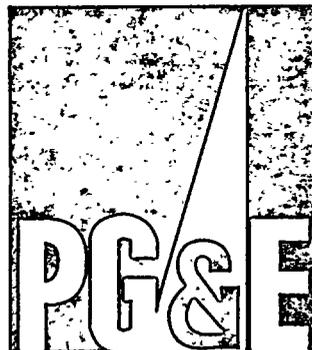
TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-54	The LCO, Required Actions, and Surveillances are revised per the DCPD specific CTS to incorporate Region 2 fuel storage requirements.	Yes, per LA 116/114.	No	No	No
3.7-55	NUREG-1431 Specification 3.7.14 is not used since an equivalent safety grade system does not exist. Therefore, the deletion is per the current licensing basis.	Yes	No	No	No
3.7-56	This change creates a new SR for the MSIVs [and MFIVs] to distinguish between the IST and the automatic actuation testing of these isolation valves. The SR allows credit for an actual actuation, if one occurs, to satisfy the surveillance requirements. Although SRs 3.7.2.2 and 3.7.3.2 are new SRs, they may be performed in conjunction with SRs 3.7.2.1 and 3.7.3.1. Therefore, the Note allowing testing to be performed on MODE 3 is also needed for these new SRs.	Yes	Yes	Yes	Yes
3.7-57	This change establishes appropriate Required Actions and Completion Times for ventilation system pressure envelope degradation.	{No, retained CTS.}	Yes	Yes	Yes

JLS Conversion to Improved Technical Specifications Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 3/4.8 - Electrical Power Systems

ITS 3.8 - Electrical Power Systems



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.8.

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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(5 Pages)
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CROSS-REFERENCE TABLE FOR 3.8
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.8.1.1	LCO	a.		3.8.1	LCO	a.	
3.8.1.1	LCO	b.		3.8.1	LCO	b.	3.8-01
3.8.1.1	LCO	b.1.	01-01-A	3.8.1.4	SR		
3.8.1.1	LCO	b.2.	01-01-A	3.8.1	LCO	c	3.8-04
3.8.1.1	LCO	b.2.	01-01-A 01-31-A 01-38-LG	3.8.3	LCO		3.8-10
3.8.1.1	APP			3.8.1	APP		
3.8.1.1	ACT	a.	01-03-LS5 01-31-A 01-32-M	3.8.1	ACTION	A.	3.8-02 3.8-32
3.8.1.1	ACT	b.	01-06-LS7 01-05-LS6 01-32-M 01-50-LS17	3.8.1	ACTION	B.	3.8-02 3.8-03
3.8.1.1	ACT	c.	01-08-LS8	3.8.1	ACTION	D.	3.8-32
3.8.1.1	ACT	d.1.	01-06-LS7	3.8.1	ACTION	B.	
3.8.1.1	ACT	d.2		3.8.1	ACTION	B.	3.8-43
3.8.1.1	ACT	e.	01-03-LS5 01-10-M 01-22-M	3.8.1	ACTION	C.	
3.8.1.1	ACT	f.		3.8.1	ACTION	E.	3.8-01
3.8.1.1	ACT	g.	01-01-A 01-10-M	3.8.1	ACTION	F.	3.8-01 3.8-04
3.8.1.1	ACTION	h.	01-01-A	3.8.1	ACTION	G.	3.8-01 3.8-04
3.8.1.1	ACTION	a.	01-03-LS5 01-31-A 01-32-M	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	b.	01-06-LS7 01-05-LS6 01-32-M 01-50-LS17	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	c.	01-08-LS8	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	e.	01-03-LS5 01-10-M 01-22-M	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	f.		3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	g	01-01-A 01-10-M	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	h.	01-01-A	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	NEW	01-48-M	3.8.3	LCO		3.8-18
3.8.1.1	ACTION	NEW	01-49-LS16	3.8.3	ACTION	A.	3.8-10
3.8.1.1	ACTION	NEW	01-48-M	3.8.3	ACTION	B	
3.8.1.1	ACTION	NEW	01-48-M	3.8.3.2	SR		
3.8.1.1	ACTION	NEW	01-49-LS16	3.8.3	ACTION	C.	3.8-10

CROSS-REFERENCE TABLE FOR 3.8
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.8.1.1	ACTION	NEW	01-49-LS16	3.8.3	ACTION	D.	3.8-10
3.8.1.1	ACTION	NEW	01-48-M	3.8.3	ACTION	E	3.8-12
3.8.1.1	ACTION	NEW	01-48-A	3.8.3.4	SR		3.8-12
3.8.1.1	ACTION	NEW	01-64-M	3.8.3	ACTION	F	3.8-18
3.8.1.1	ACTION	NEW	01-64-M	3.8.3.6	SR		3.8-18
3.8.1.1	ACTION	NEW	01-48-M 01-64-M	3.8.3	ACTION	G.	3.8-10 3.8-18
3.8.1.1	ACTION	NEW	01-48-M	3.8.3	ACTION	H	3.8-10
4.8.1.1.1a.	SR			3.8.1.1	SR		
4.8.1.1.1b.1.	SR		01-11-TR1	3.8.1.8	SR		3.8-06 3.8-20
4.8.1.1.1b.2.	SR		01-11-TR1	3.8.1.12	SR	d.	3.8-06 3.8-20
4.8.1.1.1b.2.	SR			3.8.1.12	SR	e.	3.8-06 3.8-20
4.8.1.1.2	SR		01-13-LS1 01-15-A 01-40-A	3.8.1.2	SR		3.8-05
4.8.1.1.2	SR		01-12-A 01-13-LS1 01-16-LG 01-40-A	3.8.1.7	SR		
4.8.1.1.2	SR		01-13-LS1 01-17-LS10 01-18-LS11 01-19-LS12 01-40-A 01-53-M	3.8.1.3	SR		3.8-05
4.8.1.1.2a.1)	SR		01-01-A	3.8.1.4	SR		
4.8.1.1.2a.4)	SR		01-20-LG			Not Used	
4.8.1.1.2a.5)	SR		01-20-LG			Not Used	
4.8.1.1.2b.1)	SR		01-20-LG			Not Used	
4.8.1.1.2b.2)	SR		01-20-LG 01-55-M	3.8.1.18	SR		3.8-09 3.8-20
4.8.1.1.2b.3)	SR		01-23-LG 01-24-LS13 01-25-M 01-26-M	3.8.1.9	SR		
4.8.1.1.2b.4)	SR		01-18-LS11 01-26-M 01-27-LS9	3.8.1.10	SR		3.8-20
4.8.1.1.2b.5)	SR		01-11-TR1 01-15-A 01-55-M	3.8.1.11	SR		3.8-20
4.8.1.1.2b.6)	SR		01-11-TR1	3.8.1.12	SR	a.	3.8-20
4.8.1.1.2b.6)	SR		01-15-A	3.8.1.12	SR	b.	3.8-20

CROSS-REFERENCE TABLE FOR 3.8
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
4.8.1.1.2b.6)	SR			3.8.1.12	SR	c.	3.8-20
4.8.1.1.2b.7)	SR		01-11-TR1 01-55-M	3.8.1.19	SR		3.8-09 3.8-20
4.8.1.1.2b.7)	SR	a.		3.8.1.19	SR		
4.8.1.1.2b.7)	SR	b.	01-15-A	3.8.1.19	SR		
4.8.1.1.2b.7)	SR	c.	01-29-A	3.8.1.13	SR		3.8-20
4.8.1.1.2b.8)	SR		01-18-LS11 01-26-M 01-30-LG 01-31-A	3.8.1.14	SR		3.8-20
4.8.1.1.2b.9)	SR		01-20-LG			Not Used	
4.8.1.1.2b.10)	SR		01-14-M	3.8.1.16	SR		3.8-20
4.8.1.1.2b.11)	SR		01-11-TR1 01-55-M	3.8.1.17	SR		3.8-08 3.8-20
4.8.1.1.2b.12)	SR		01-20-LG			Not Used	
4.8.1.1.2c	SR		01-07-LS3 01-15-A 01-35-M 01-66-TR3	3.8.1.20	SR		3.8-40
4.8.1.1.2d	SR		01-33-LS15	3.8.1.5	SR		
4.8.1.1.2e	SR		01-18-LS11 01-31-A 01-19-LS-12	3.8.1.15	SR		3.8-40
4.8.1.1.3a.1)	SR		01-01-A	3.8.3.1	SR		3.8-10
4.8.1.1.3a.2)	SR			3.8.1.6	SR		
4.8.1.1.3b	SR		01-01-A	3.8.3.5	SR		
4.8.1.1.3c	SR		01-37-LG	3.8.3.3	SR		
4.8.1.1.3d	SR		01-37-LG	3.8.3.3	SR		
4.8.1.1.3e.1)	SR		01-38-LG			Not Used	3.8-31
4.8.1.1.3e.2)	SR		01-39-LG			Not Used	
4.8.1.1.4	SR		01-13-LS1	5.6.7		Not Used	3.8-05
Table 4.8-1	TBL		01-13-LS-1	3.8.1-1	TBL	Not Used	3.8-05
Table 4.8-2a	TBL		01-20-LG			Not Used	
Table 4.8-2b	TBL		01-20-LG			Not Used	
3.8.1.2	LCO	a.	01-41-M	3.8.2	LCO	a.	
3.8.1.2	LCO	b.	01-01-A	3.8.2	LCO	b, c	3.8-04 3.8-15
3.8.1.2	LCO		01-48-M 01-64-M	3.8.3	APP		
3.8.1.2	LCO	b.1.	01-01-A	3.8.2.1	SR		3.8-17
3.8.1.2	LCO	b.2.	01-01-A 01-31-A	3.8.3	LCO		3.8-10 3.8-11
3.8.1.2	APP		01-42-M	3.8.2	APP		

CROSS-REFERENCE TABLE FOR 3.8
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.8.1.2	ACTION		01-43-M 01-44-LG 01-46-A 01-59-LS20	3.8.2	ACTION	A	3.8-15 3.8-35
3.8.1.2	ACTION		01-45-M	3.8.2	ACTION	B	3.8-04 3.8-35
4.8.1.2	SR		01-47-LS4	3.8.2.1	SR		3.8-17
3.8.2.1	LCO		03-01-LG	3.8.9	LCO		3.8-15
3.8.2.1	LCO		03-02-A	3.8.7	LCO		3.8-15
3.8.2.1	APP			3.8.7	APP		
3.8.2.1	ACT	a.	03-01-LG 03-04-M	3.8.9	ACTION	A.	
3.8.2.1	ACT	b.	03-01-LG 03-04-M	3.8.9	ACTION	B.	3.8-15
3.8.2.1	ACT	c.		3.8.4	ACTION	B.	3.8-13
3.8.2.1	ACT	d.	03-01-LG 03-04-M	3.8.9	ACTION	C.	
3.8.2.1	ACT	NEW	03-05-A	3.8.9	ACTION	E.	
4.8.2.1	SR			3.8.9.1	SR		
3.8.2.2	LCO	b	03-06-M	3.8.8	LCO		
3.8.2.2	LCO		03-06-M	3.8.10	LCO		3.8-15
3.8.2.2	APP		01-42-M	3.8.8	APP		
3.8.2.2	APP		01-42-M	3.8.10	APP		3.8-29
3.8.2.2	ACT		03-03-LS20 03-07-M	3.8.10	ACTION	A.	3.8-15 3.8-35
4.8.2.2	SR			3.8.8.1	SR		
4.8.2.2	SR			3.8.10.1	SR		3.8-15
3.8.3.1	LCO		02-01-LG	3.8.4	LCO		3.8-13
3.8.3.1	APP			3.8.4	APP		
3.8.3.1	APP			3.8.6	APP		
3.8.3.1	ACT		02-01-LG	3.8.4	ACTION	A.	
3.8.3.1	ACT			3.8.4	ACTION	C.	
4.8.3.1a.1)	SR		02-06-LS22	3.8.6.1	SR		3.8-34
4.8.3.1a.2)	SR		02-06-LS22	3.8.4.1	SR		3.8-34
4.8.3.1b.1)	SR		02-17-LS2	3.8.6.2	SR		3.8-14
4.8.3.1b.2)	SR		02-23-A 02-24-LG	3.8.4.2	SR		
4.8.3.1b.3)	SR		02-18-LG 02-22-A	3.8.6.3	SR		
4.8.3.1c.1)	SR		02-02-A	3.8.4.3	SR		
4.8.3.1c.2)	SR		02-03-M	3.8.4.4	SR		

CROSS-REFERENCE TABLE FOR 3.8
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
4.8.3.1c.3)	SR		02-23-A	3.8.4.5	SR		
4.8.3.1c.4)	SR			3.8.4.6	SR		3.8-20 3.8-39
4.8.3.1d.	SR		02-04-M	3.8.4.7	SR		3.8-20 3.8-38
4.8.3.1e.	SR		02-04-M 02-05-M	3.8.4.8	SR		3.8-16
4.8.3.1f.	SR		02-04-M 02-05-M 02-18-LG 02-21-LS14	3.8.4.8	SR		3.8-16
Table 4.8-3	TBL		02-14-A	3.8.6-1	TBL		
Table 4.8-3	TBL	Note 1	02-09-M	3.8.6	LCO		3.8-13
Table 4.8-3	TBL	Note 1	02-08-LS21 02-09-M	3.8.6	ACTION	A.	
Table 4.8-3	TBL	Note 2	02-08-LS21			Not Used	
Table 4.8-3	TBL	Note 3		3.8.6	ACTION	B.	
Table 4.8-3	TBL	Note 4	02-10-A	3.8.6-8	TBL	Note b	
Table 4.8-3	TBL	Note 5	02-11-A	3.8.6-1	TBL	Note c	
Table 4.8-3	TBL	Note 6	02-12-LG			Not Used	
Table 4.8-3	TBL	NEW	02-07-A	3.8.6-1	TBL	Note a	
3.8.3.2	LCO		02-01-LG 02-13-M	3.8.5	LCO		3.8-13
3.8.3.2	APP		01-42-M	3.8.5	APP		
3.8.3.2	ACT		02-01-LG 02-19-LS20	3.8.5	ACTION	A.	3.8-35
4.8.3.2	SR		02-01-LG 02-15-LS4	3.8.5.1	SR		
3.8.4.1	LCO		04-01-R			Not Used	
3.8.4.1	APP		04-01-R			Not Used	
4.8.4.1	SR		04-01-R			Not Used	
3.8.4.2	LCO		04-01-R			Not Used	
3.8.4.2	APP		04-01-R			Not Used	
4.8.4.2	SR		04-01-R			Not Used	

CROSS-REFERENCE TABLE FOR 3.8
Sorted by Improved TS

<u>Current TS</u>				<u>Improved TS</u>			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.8.1.1	LCO	a.		3.8.1	LCO	a.	
3.8.1.1	LCO	b.		3.8.1	LCO	b.	3.8-01
3.8.1.1	LCO	b.2.	01-01-A	3.8.1	LCO	c.	3.8-04
3.8.1.1	APP			3.8.1	APP		
3.8.1.1	ACTION	a.	01-03-LS5 01-31-A 01-32-M	3.8.1	ACTION	A.	3.8-02 3.8-32
3.8.1.1	ACTION	b.	01-06-LS7 01-05-LS6 01-32-M 01-50-LS17	3.8.1	ACTION	B.	3.8-02 3.8-03
3.8.1.1	ACTION	d.1.	01-06-LS7	3.8.1	ACTION	B.	
3.8.1.1	ACTION	d.2		3.8.1	ACTION	B.	3.8-43
3.8.1.1	ACTION	e.	01-03-LS5 01-10-M 01-22-M	3.8.1	ACTION	C.	
3.8.1.1	ACTION	c.	01-08-LS8	3.8.1	ACTION	D.	3.8-32
3.8.1.1	ACTION	f.		3.8.1	ACTION	E.	3.8-01
3.8.1.1	ACTION	g.	01-01-A 01-10-M	3.8.1	ACTION	F.	3.8-01 3.8-04
3.8.1.1	ACTION	h.	01-01-A	3.8.1	ACTION	G.	3.8-01 3.8-04
3.8.1.1	ACTION	a.	01-03-LS5 01-31-A 01-32-M	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	b.	01-06-LS7 01-05-LS6 01-32-M 01-50-LS17	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	c.	01-08-LS8	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	e.	01-03-LS5 01-10-M 01-22-M	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	f.		3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	g.	01-01-A 01-10-M	3.8.1	ACTION	H.	3.8-04
3.8.1.1	ACTION	h	01-01-A	3.8.1	ACTION	H.	3.8-04
		NEW		3.8.1	ACTION	I.	3.8-01
		NEW		3.8.1	ACTION	J.	3.8-01
4.8.1.1.1a.	SR			3.8.1.1	SR		
4.8.1.1.2	SR		01-13-LS1 01-15-A 01-40-A	3.8.1.2	SR		3.8-05 3.8-40

CROSS-REFERENCE TABLE FOR 3.8
Sorted by Improved TS

<u>Current TS</u>				<u>Improved TS</u>			
Item	Code	Para.	Change	Item	Code	Para.	Change
4.8.1.1.2	SR		01-13-LS1 01-17-LS10 01-18-LS11 01-19-LS12 01-40-A 01-53-M	3.8.1.3	SR		3.8-05
3.8.1.1	LCO	b.1	01-01-A	3.8.1.4	SR		
4.8.1.1.2a.1)	SR		01-01-A	3.8.1.4	SR		
4.8.1.1.2d	SR		01-33-LS15	3.8.1.5	SR		
4.8.1.1.3a.2)	SR			3.8.1.6	SR		
4.8.1.1.2	SR		01-12-A 01-13-LS1 01-16-LG 01-40-A	3.8.1.7	SR		
4.8.1.1.1b.1.	SR		01-11-TR1	3.8.1.8	SR		3.8-06 3.8-20
4.8.1.1.2b.3)	SR		01-11-TR1 01-23-LG 01-24-LS13 01-25-M 01-26-M	3.8.1.9	SR		
4.8.1.1.2b.4)	SR		01-11-TR1 01-18-LS11 01-26-M 01-27-LS9	3.8.1.10	SR		3.8-20
4.8.1.1.2b.5)	SR		01-11-TR1 01-15-A 01-55-M	3.8.1.11	SR		3.8-20
4.8.1.1.2b.6)	SR		01-11-TR1	3.8.1.12	SR	a.	3.8-20
4.8.1.1.2b.6)	SR		01-15-A	3.8.1.12	SR	b.	3.8-20
4.8.1.1.2b.6)	SR			3.8.1.12	SR	c.	3.8-20
4.8.1.1.1b.2	SR		01-11-TR1	3.8.1.12	SR	d.	3.8-06 3.8-20
4.8.1.1.1b.2	SR			3.8.1.12	SR	e.	3.8-06 3.8-20
4.8.1.1.2b.7)	SR	c.	01-29-A	3.8.1.13	SR		3.8-20
4.8.1.1.2b.8)	SR		01-11-TR1 01-18-LS11 01-26-M 01-30-LG 01-31-A	3.8.1.14	SR		3.8-20
4.8.1.1.2e.	SR		01-18-LS11 01-31-A 01-19-LS-12	3.8.1.15	SR		3.8-40
4.8.1.1.2b.10)	SR		01-11-TR1	3.8.1.16	SR		3.8-20
4.8.1.1.2b.11)	SR		01-11-TR1 01-55-M	3.8.1.17	SR		3.8-08 3.8-20

CROSS-REFERENCE TABLE FOR 3.8
Sorted by Improved TS

<u>Current TS</u>				<u>Improved TS</u>			
Item	Code	Para.	Change	Item	Code	Para.	Change
4.8.1.1.2b.2)	SR		01-11-TR1 01-20-LG 01-55-M	3.8.1.18	SR		3.8-09 3.8-20
4.8.1.1.2b.7)	SR		01-11-TR1 01-55-M	3.8.1.19	SR		3.8-09 3.8-20
4.8.1.1.2b.7)	SR	a.		3.8.1.19	SR		3.8-20
4.8.1.1.2b.7)	SR	b.	01-15-A	3.8.1.19	SR		3.8-20
4.8.1.1.2c.	SR		01-07-LS3 01-15-A 01-34-LG 01-35-M	3.8.1.20	SR		3.8-40
Table 4.8-1	TBL		01-13-LS1	3.8.1-1	TBL	Not Used	3.8-05
3.8.1.2	LCO	a.	01-41-M	3.8.2	LCO	a.	
3.8.1.2	LCO	b.	01-01-A	3.8.2	LCO	b, c.	3.8-04 3.8-15
3.8.1.2	APP		01-42-M	3.8.2	APP		
3.8.1.2	ACTION		01-44-LG 01-45-M 01-46-A 01-59-LS20	3.8.2	ACTION	A.	3.8-15 3.8-35
3.8.1.2	ACTION		01-45-M	3.8.2	ACTION	B.	3.8-04 3.8-35
3.8.1.2	LCO	b.1	01-01-A	3.8.2.1	SR		3.8-17
4.8.1.2	SR		01-47-LS4	3.8.2.1	SR		3.8-17
3.8.1.1	LCO	b.2	01-01-A 01-31-A 01-38-LG	3.8.3	LCO		3.8-10
3.8.1.2	LCO	b.2.	01-01-A 01-31-A	3.8.3	LCO		3.8-10 3.8-11
3.8.1.1	ACTION	NEW	01-48-M	3.8.3	LCO		3.8-18
3.8.1.1	LCO		01-48-M 01-64-M	3.8.3	APP		
3.8.1.1	ACTION	NEW	01-49-LS16	3.8.3	ACTION	A	3.8-10
3.8.1.1	ACTION	NEW	01-48-M	3.8.3	ACTION	B.	
3.8.1.1	ACTION	NEW	01-49-LS16	3.8.3	ACTION	C.	3.8-10
3.8.1.1	ACTION	NEW	01-49-LS16	3.8.3	ACTION	D.	3.8-10
3.8.1.1	ACTION	NEW	01-48-M	3.8.3	ACTION	E.	3.8-12
3.8.1.1	ACTION	NEW	01-64-M	3.8.3	ACTION	F.	3.8-18
3.8.1.1	ACTION	NEW	01-48-M 01-64-M	3.8.3	ACTION	G.	3.8-10 3.8-18
3.8.1.1	ACTION	NEW	01-48-M	3.8.3	ACTION	H.	3.8-10
4.8.1.1.3a.1)	SR		01-01-A	3.8.3.1	SR		3.8-10

CROSS-REFERENCE TABLE FOR 3.8
Sorted by Improved TS

<u>Current TS</u>				<u>Improved TS</u>			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.8.1.1	ACTION	NEW	01-48-M	3.8.3.2	SR		
4.8.1.1.3c	SR		01-37-LG	3.8.3.3	SR		
4.8.1.1.3d	SR		01-37-LG	3.8.3.3	SR		
3.8.1.1	ACTION	NEW	01-48-M	3.8.3.4	SR		3.8-12
4.8.1.1.3b	SR		01-01-A	3.8.3.5	SR		
4.8.1.1.3e.1)	SR		01-38-LG	3.8.3.6	SR	Not Used	3.8-31
3.8.1.1	ACTION	NEW	01-64-M	3.8.3.6	SR		3.8-18
3.8.3.1	LCO		02-01-LG	3.8.4	LCO		3.8-13
3.8.3.1	APP			3.8.4	APP		
3.8.3.1	ACTION		02-01-LG	3.8.4	ACTION	A	
3.8.2.1	ACTION	c.		3.8.4	ACTION	B	3.8-13
3.8.2.1	ACTION			3.8.4	ACTION	C	
4.8.3.1a.2)	SR		02-06-LS22	3.8.4.1	SR		3.8-34
4.8.3.1b.2)	SR		02-23-A 02-24-LG	3.8.4.2	SR		
4.8.3.1c.1)	SR		02-02-A	3.8.4.3	SR		3.8-41
4.8.3.1c.2)	SR		02-03-M	3.8.4.4	SR		
4.8.3.1c.3)	SR		0-23-A	3.8.4.5	SR		
4.8.3.1c.4)	SR			3.8.4.6	SR		3.8-20 3.8-39
4.8.3.1d.	SR		02-04-M	3.8.4.7	SR		3.8-20 3.8-38
4.8.3.1e.	SR		02-04-M 02-05-M	3.8.4.8	SR		3.8-16 3.8-20
4.8.3.1f.	SR		02-04-M 02-05-M 02-18-LG 02-21-LS14	3.8.4.8	SR		3.8-16 3.8-20
3.8.3.2	LCO		02-01-LG 02-13-M	3.8.5	LCO		3.8-13
3.8.3.2	APP		01-42-M	3.8.5	APP		
3.8.3.2	ACTION		02-01-LG 02-19-LS20	3.8.5	ACTION	A	3.8-35
4.8.3.2	SR		02-01-LG 02-15-LS4	3.8.5.1	SR		
Table 4.8-3	TBL	Note 1	02-09-M	3.8.6	LCO		3.8-13
3.8.3.1	APP			3.8.6	APP		
Table 4.8-3	TBL	Note 1	02-08-LS21 02-09-M	3.8.6	ACTION	A	
Table 4.8-3	TBL	Note 3		3.8.6	ACTION	B	

CROSS-REFERENCE TABLE FOR 3.8
Sorted by Improved TS

<u>Current TS</u>				<u>Improved TS</u>			
Item	Code	Para.	Change	Item	Code	Para.	Change
4.8.3.1a.1)	SR		02-06-LS22	3.8.6.1	SR		3.8-34
4.8.3.1b.1)	SR		02-17-LS2	3.8.6.2	SR		3.8-14
4.8.3.1b.3)	SR			3.8.6.3	SR		
Table 4.8-3	TBL		02-14-A	3.8.6-1	TBL		
Table 4.8-3	TBL	NEW	02-07-A	3.8.6-1	TBL	Note a	
Table 4.8-3	TBL	Note 4	02-10-A	3.8.6-1	TBL	Note b	
Table 4.8-3	TBL	Note 5	02-11-A	3.8.6-1	TBL	Note c	
3.8.2.1	LCO		03-02-A	3.8.7	LCO		3.8-15
3.8.2.1	APP			3.8.7	APP		
3.8.2.1	ACTION	b.	03-01-LG 03-04-M	3.8.7	ACTION	A	3.8-15
3.8.2.1	ACTION	b.	03-01-LG 03-04-M	3.8.7	ACTION	B	
4.8.2.1.	SR			3.8.7.1	SR		
3.8.2.2	LCO	b	03-06-M	3.8.8	LCO		3.8-15
3.8.2.2	APP		01-42-M	3.8.8	APP		
3.8.2.2	ACTION		03-03-LS20 03-07-M	3.8.8	ACTION	A	3.8-35
4.8.2.2	SR			3.8.8.1	SR		
3.8.2.1	LCO		03-01-LG	3.8.9	LCO		3.8-15
3.8.2.1	APP			3.8.9	APP		
3.8.2.1	ACTION	a.	03-01-LG 03-04-M	3.8.9	ACTION	A.	
3.8.2.1	ACTION	b.	03-01-LG 03-04-M	3.8.9	ACTION	B.	3.8-15
3.8.2.1	ACTION	d.	03-01-LG 03-04-M	3.8.9	ACTION	C.	
3.8.2.1	ACTION	NEW	03-05-A	3.8.9	ACTION	E.	3.8-15
4.8.2.1	SR			3.8.9.1	SR		
3.8.2.2	LCO		03-06-M	3.8.10	LCO		3.8-15
3.8.2.2	APP		01-42-M	3.8.10	APP		3.8-29
3.8.2.2	ACTION		03-03-LS20 03-07-M	3.8.10	ACTION	A.	3.8-15 3.8-35
4.8.2.2	SR			3.8.10.1	SR		3.8-15

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

**Methodology for Cross-Reference Tables
(Continued)**

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
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Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).

Note: The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS; separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
3.8.1.1	3/4 8-1
3.8.1.2	3/4 8-11
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3.8.2.2	3/4 8-14
3.8.3.1	3/4 8-15
3.8.3.2	3/4 8-18
3.8.4.1	3/4 8-19
3.8.4.2	3/4 8-20

Methodology (2 Pages)

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

- a. Two independent circuits (one with delayed access) between the offsite transmission network and the Onsite Class 1E Distribution System, and
- b. Three separate and independent diesel generators, each with:
 - 1. A separate engine-mounted fuel tank containing a minimum volume of 250 gallons of fuel, and
 - 2. Two supply trains of the Diesel Fuel Oil Storage and Transfer System containing a minimum combined storage of 33,000 gallons of fuel for one unit operation* and 65,000 gallons of fuel for two unit operation.**

01-01-A
01-31-A
01-48-M
01-48-M
01-64-M

(new) A lube oil inventory shall be within limits.

(new) A starting air receiver pressure shall be within limits, and

(new) A turbocharger air assist air receiver pressure shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

- a. With one offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specification 4.8.1.1.a. within 1 hour and at least once per 8 hours thereafter. If each of the diesel generators have not been successfully tested within the past 24 hours demonstrate its OPERABILITY by performing Specification 4.8.1.1.2a.2) separately for each such diesel generator within 24 hours. Restore the offsite circuit to OPERABLE status within 72*** hours and within 10 days from discovery of failure to meet the LCO or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

.01-03-LS5
01-31-A
01-32-M

* The performance of diesel fuel oil tank cleaning Technical Specification Surveillance Requirement 4.8.1.1.3.e requires one fuel oil storage tank to be removed from service to be drained and cleaned. During this surveillance, the diesel generator fuel oil storage requirement for one unit operation in Modes 1 through 4 and one unit operation in Mode 6 with at least 23 feet of water above the reactor vessel flange or with the reactor vessel defueled is 35,000 gallons. The tank being cleaned may be inoperable for up to 10 days. For the duration of tank cleaning, temporary onsite fuel oil storage of 24,000 gallons will be maintained. Prior to removal of a tank from service, the offsite circuits required by Technical Specification 3.8.1.1.a will be verified to be OPERABLE.

01-38-R
01-01-A

** The performance of modifications to the diesel fuel oil storage and transfer system requires one fuel oil storage tank at a time to be drained and replaced with a new storage tank. During this period, the diesel generator fuel oil storage requirement for two unit operation in Modes 1, 4, or for one unit operation in Modes 1, 4 and one unit in Mode 5 or 6 is 35,000 gallons. A total of up to 120 days may be required to complete the replacement of both tanks. For the duration of the tank replacement, temporary onsite storage of 30,000 gallons will be maintained. Prior to removal of a tank from service, the offsite circuits required by Technical Specification 3.8.1.1.a. will be verified to be OPERABLE.

01-31-A

*** For Unit 1 Cycle 8, the allowed outage time may be extended to 120 hours on a one time basis for installation of auxiliary transformer 11.

01-31-A

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. With a diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the A.C. offsite sources by performing Specification 4.8.1.1.1a within 1 hour and at least once per 8 hours thereafter; and declare required feature supported by the inoperable diesel generator inoperable when its required redundant feature is inoperable within 4 hours from discovery of diesel generator inoperability concurrent with inoperability of redundant required feature; and determine the OPERABLE diesel generators are not inoperable due to a common cause failure or and if the diesel generator became inoperable due to any cause other than preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators by performing Specification 4.8.1.1.2a.2) within 24 hours*; restore the diesel generator to OPERABLE status within 7 days and within 10 days from discovery of failure to meet the LCO or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

01-06-LS7

01-50-LS17

01-05-LS6

01-32-M

c. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, in addition to the requirements of ACTIONS a. and b. above demonstrate the OPERABILITY of the remaining A.C. sources by performing Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; and if the diesel generator became inoperable due to any cause other than preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators by performing Specification 4.8.1.1.2a.2) within 8 hours; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with ACTION a. or b., as appropriate with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable A.C. power source. A successful test of diesel OPERABILITY per Specification 4.8.1.1.2a.2) performed under this ACTION statement for OPERABLE diesels or a restored to OPERABLE diesel satisfies the diesel generator test requirement of ACTION a. or b.

01-08-LS8

d. With one diesel generator inoperable in addition to ACTION b. or c. above verify that:

1. All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generators as a source of emergency power are also OPERABLE, and
2. When in MODE 1, 2, or 3 that at least two auxiliary feedwater pumps are OPERABLE.

01-08-LS8

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

01-06-LS7

*This test is required to be completed regardless of when the inoperable diesel generator is restored to operability.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

- e. With two of the above required offsite A.C. circuits inoperable, ~~within 12 hours from the discovery of two offsite circuits inoperable concurrent with inoperability of redundant required feature(s) declare required feature(s) inoperable when its redundant required feature(s) is inoperable; and demonstrate the OPERABILITY of three diesel generators by performing the requirements of Specification 4.8.1.1.2a.2) within 8 hours, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~ 01-22-M
~~Following restoration of one offsite source, follow ACTION a. with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable offsite A.C. circuit. A successful test(s) of diesel generator OPERABILITY per Specification 4.8.1.1.2a.2) performed under this ACTION statement for the OPERABLE diesel generators satisfies the diesel generator test requirement of ACTION a.~~ 01-03-LS5
01-10-M
01-03-LS5
- f. With two or more of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; ensure at least two of the diesel generators are OPERABLE within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. With two diesel generators OPERABLE follow ACTION b. with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable diesel generator. A successful test of diesel generator OPERABILITY per Specification 4.8.1.1.2a.2) performed under this ACTION statement for a restored to OPERABLE diesel generator satisfies the diesel generator test requirement of ACTION b.
- g. With one supply train of the Diesel Fuel Oil Storage and Transfer System inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and be in ~~HOT~~ COLD SHUTDOWN within the following 6 ~~30~~ hours. 01-01-A
01-10-M
- h. With both supply trains of the Diesel Fuel Oil Storage and Transfer System inoperable, restore at least one supply train, ~~including the common storage system,~~ to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours. 01-01-A

ACTION (Continued)

- (new) With combined fuel oil in storage tanks not within limits, verify combined fuel oil level is at least 29,000 gallons for each unit operating in MODES 1, 2, 3, or 4, and at least 23,000 gallons for each unit operating in MODES 5 or 6 and restore fuel oil to within limits in 48 hours, or immediately declare associated diesel generators inoperable and if the associated unit is in MODES 1, 2, 3, or 4, be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. 01-49-LS16
- (new) With both units in MODES 1, 2, 3, or 4 and a combined lube oil inventory less than 650 gallons but greater than 610 gallons, restore the lube oil inventory to greater than 650 gallons or with one unit in MODE 1, 2, 3, or 4 and the other unit in MODES 5 or 6 and a combined lube oil inventory of less than 590 gallons but greater than 520 gallons, restore the lube oil inventory to greater than 590 gallons in 48 hours, or immediately declare associated diesel generators inoperable and if the associated unit is in MODES 1, 2, 3, or 4, be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. 01-48-M
- (new) With one or more diesel generators with starting air receiver pressures less than 180 psig but greater than 150 psig, restore at least one starting air receiver pressure per diesel generator to greater than 180 psig in 48 hours. 01-48-M
- (new) With one or more diesel generators with turbocharger air assist air receiver pressure less than 180 psig but greater than 150 psig, restore air receiver pressure to greater than 180 psig in 48 hours. 01-64-M
- (new) With one or more fuel oil storage tanks with stored fuel oil total particulates not within limits, restore fuel oil total particulates to within limits in 7 days, or immediately declare associated diesel generators inoperable and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. 01-49-LS16
- (new) With one or more fuel oil storage tanks with new fuel oil properties not within limits, restore stored fuel oil properties to within limits in 30 days, or immediately declare associated diesel generators inoperable and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. 01-49-LS16
- (new) With starting air receiver pressure, or turbocharger air assist air receiver pressure not within limits for reasons other than above, immediately declare associated diesel generator inoperable. 01-48-M
01-64-M

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

- a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and
- b. Demonstrated OPERABLE at least once per 18 months during shutdown by:
 - 1) Transferring 4 kV vital bus power supply from the normal circuit to the alternate circuit (manually and automatically) and to the delayed access circuit (manually), and
 - 2) Verifying that on a Safety Injection ~~actual~~ or test signal, without loss of offsite power, the preferred, immediate access offsite power source energizes the emergency busses with permanently connected loads and energizes the auto-connected emergency (accident) loads through sequencing timers. 01-11-TR1

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE*:

- a. ~~In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS At Least once per 31 days by:~~ 01-12-A
 - 1) ~~Verifying the fuel level in the engine-mounted fuel tank contains > 250 gallons of fuel oil.~~ 01-40-LS2
01-13-LS1
01-17-LS10
 - 2) Verifying the diesel starts from ~~standby~~ ambient condition and accelerates to at least 900 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4160 + 240/-375 volts and 60 ± 1.2 Hz within 13 seconds after the start signal. ~~The diesel generator shall be started for this test by using one of the following signals:~~
 - a) ~~Manual, or~~ 01-01-A
 - b) ~~Simulated loss of offsite power by itself (Startup bus undervoltage), or~~ 01-15-A
 - c) ~~A Safety Injection actuation test signal by itself.~~ 01-16-LG

* All diesel generator starts for the purpose of this surveillance test may be preceded by an engine prelube period. Further, all surveillance tests, with the exception of once per 184 days, may also be preceded by warmup procedures (e.g., gradual acceleration and/or gradual loading > 150 sec) as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized. 01-12-A

** ~~Diesel generator loading may include gradual loading as recommended by the manufacturer, including a warmup period prior to loading.~~ 01-17-LS10

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

	01-53-M
3) Verifying the generator *** is synchronized, loaded to greater than or equal to 2484 kW in less than or equal to 60 seconds, and operates for greater than or equal to 60 minutes at a load ≥ 2370 kW and ≤ 2610 kW .	01-17-LS10 01-18-LS11 01-19-LS12
4) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses, and	01-20-LG
5) Verifying the diesel engine protective relay trip cutout switch is returned to the cutout position following each diesel generator test.	01-20-LG
(new) Verifying lube oil inventory is at least 650 gallons	01-48-M
(new) Verifying each diesel generator has at least one starting air receiver with a pressure at least 180 psig, and	01-48-M
(new) Verifying each DG turbocharger air assist air receiver pressure is at least 180 psig	01-64-M
b. At least once per 18 months during shutdown, by:	
1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;	01-20-LG 01-55-M
2) Verifying # that the load sequence timers are OPERABLE with each load sequence timer within the its limits specified in Table 4.8-2;	01-20-LG
3) Verifying the generator capability to reject a load of greater than or equal to 508 kW while maintaining voltage at 4160 ± 240/ 375 volts and frequency at 60 ± 3 Hz; Verifying each DG rejects a load** greater than or equal to its single largest post-accident load, and	01-26-M 01-23-LG
a. Following load rejection, the frequency is ≤ 63 Hz	
b. Within 2.4 seconds following load rejection, the voltage is ≥ 3785 V and ≤ 4400 V; and	01-24-LS13
c. Within 2.4 seconds following load rejection, the frequency is ≥ 58.8 Hz and ≤ 61.2 Hz	01-25-M
4) Verifying the generator capability to reject a load of greater than or equal to 2370 kW and ≤ 2610 kW while operating at a power factor of ≤ 0.87, 2484 kW without tripping. The generator voltage shall not exceed 6200 4580 volts during and following the load rejection;	01-18-LS11 01-26-M 01-27-LS9

- 5) Verify by Simulating or on an actual signal # a loss of offsite power by itself, and: 01-11-TR1
- a) Verifying de-energization of the emergency busses and load shedding from the emergency busses, and 01-55-M
- b) Verifying the diesel starts from standby conditions on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the required auto-connected loads through sequencing timers and operates for greater than or equal to 5 minutes while its generator is loaded with the permanent and auto-connected loads. After energization of these loads, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 + 240/-375 volts and 60 ± 1.2 Hz during this test. 01-15-A

Momentary transients outside the load range do not invalidate this test. 01-19-LS12

** If performed with the DG synchronized with offsite power, this Surveillance shall be performed at a power factor of ≥ 0.9 01-26-M

*** This surveillance shall be conducted on only one DG at a time 01-53-M

This surveillance shall not be performed in MODE 1, 2, 3, or 4. 01-55-M

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- 6) Verifying that on a Safety Injection actual or test signal without loss of offsite power, the diesel generator starts from standby conditions on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 + 240/-375 volts and 60 + 1.2 Hz within 13 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test; 01-11-TR1
01-15-A

 - 7) Verify by Simulating# or by an actual a loss of offsite power in conjunction with a Safety Injection actual or test signal, and: 01-55-M
01-11-TR1
 - a) Verifying de-energization of the emergency busses and load shedding from the emergency busses;
 - b) Verifying the diesel starts from standby conditions on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through sequencing timers and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization of these loads, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 + 240/-375 volts and 60 ± 1.2 Hz during this test; and 01-15-A
 - e) Verifying that all automatic diesel generator trips, except engine overspeed, low lube oil pressure and generator differential, are bypassed when the diesel engine trip cutout switch is in the cutout position and the diesel is aligned for automatic operation. 01-29-A
01-26-M

 - 8) Verifying the diesel generator operates at a power factor $\leq 0.87^{**}$ for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 2750 kW, 2625 kW and ≤ 2890 kW* and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 2484, 2370 kW and ≤ 2610 kW*. The generator voltage and frequency shall be 4160 + 240/-375 volts and 60 + 1.2 Hz within 13 seconds after the start signal. For Units 1 and 2 Cycle 7: Within 5 minutes after completing this 24 hour test, perform Specification 4.8.1.1.2b.5)b);* 01-18-LS11
01-19-LS12
01-30-LG
01-31-A

 - 9) Verifying that the auto connected loads to each diesel generator do not exceed the maximum rating of 2750 kW; 01-20-LG

 - 10) Verifying# the diesel generator's capability to: 01-55-M
- * For Units 1 and 2 Cycle 7: If Specification 4.8.1.1.2b.5)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead the diesel generator may be operated at 2484 kW for 1 hour or until operating temperature has stabilized. 01-31-A
- * Momentary transients outside the load range do not invalidate this test. 01-19-LS12
- ** Momentary transients outside the power factor range do not invalidate this test. 01-26-M
- # This surveillance shall not be performed in MODE 1, 2, 3, or 4. 01-55-M

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- a) Synchronize its isolated bus with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power.
- b) Transfer its loads to the offsite power source, and
- c) Be restored to its ~~ready to load~~ standby status.

01-14-M

- 11) Verifying# that with the diesel generator operating in a test mode, connected to its bus, a simulated or actual Safety Injection signal opens the auxiliary transformer breaker and automatically sequences the emergency loads onto the diesel generator; and

01-55-M

01-11-TR1

- ~~12) Verifying that the shutdown relay lockout feature prevents diesel generator starting only when required:~~

- ~~a) Generator differential current high, or~~
- ~~b) Engine lube oil pressure low, or~~
- ~~c) Emergency stop button actuated, or~~
- ~~d) Overspeed trip actuated.~~

01-20-LG

01-66-TR3

- c. ~~At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all diesel generators simultaneously from standby conditions, during shutdown, and verifying that all diesel generators accelerate to at least 900 rpm in less than or equal to 10 seconds and achieve voltage ≥ 3785 V and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz within 13 seconds.~~

01-15-A

01-35-M

01-07-LS3

- d. ~~At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tank.~~

01-33-LS15

- e. ~~For Units 1 and 2, Cycle 8 and after:~~

01-31-A

~~At least once per 18 months by verifying the diesel generator starts and accelerates to at least 900 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be $4160 \pm 240/-375$ volts and 60 ± 1.2 Hz within 13 seconds after the start signal. This test shall be performed within 5 minutes of shutting down the diesel generator after the diesel generator has operated for at least 2 hours at a load of greater than or equal to 2484 2370 kW and ≤ 2610 kW.~~

01-19-LS12

01-18-LS11

4.8.1.1.3 The Diesel Fuel Oil Storage and Transfer System shall be demonstrated OPERABLE:

- a. At least once per 31 days by:

- 1) Verifying the fuel level in the fuel storage tank, and
- 2) Verifying that each fuel transfer pump starts and transfers fuel from the storage system to each engine-mounted tank via installed lines.

01-01-A

b. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

01-01-A

c. By sampling new fuel oil in accordance with the Diesel Fuel Oil Testing Program ASTM D4057 prior to addition to the storage tanks and:

01-37-LG

[REDACTED]

* Momentary transients outside the load range do not invalidate this test

01-19-LS12

This surveillance shall not be performed in MODE 1, 2, 3, or 4

01-55-M

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- 1) ~~By verifying in accordance with the tests specified in ASTM D975 81 prior to addition to the storage tanks that the sample has:~~
 - a) ~~An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;~~
 - b) ~~A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification;~~
 - c) ~~A flash point equal to or greater than 125°F; and~~
 - d) ~~A clear and bright appearance with proper color when tested in accordance with ASTM D4176 82 or a water volume and sediment content of less than or equal to 0.05 volume percent when tested in accordance with ASTM D1796 83.~~
- 2) ~~By verifying within 30 and 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975 81 are met when tested in accordance with ASTM D975 81 except that the analysis for sulfur may be performed in accordance with ASTM D1552 79 or ASTM D2622 82.~~
- d. ~~Verifying stored fuel oil is tested in accordance with the Diesel Fuel Oil Testing Program. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM D2276 78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276 78, Method A;~~
- e. ~~At least once per 10 years by:~~
 - 1) ~~Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and~~
 - 2) ~~Performing a visual examination of accessible piping during an operating pressure leak test.~~

~~4.8.1.1.4 Reports All diesel generator failures, valid or non valid, shall be reported as a Special Report within 30 days to the Commission pursuant to Specification 6.9.2. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures (on a per diesel generator basis) in the last 100 valid tests is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.~~

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

<u>Number of Failures in Last 20 Valid Tests*</u>	<u>Number of Failures in Last 100 Valid Tests*</u>	<u>Test Frequency</u>
<u>≤ 1</u>	<u>≤ 5</u>	<u>At least once per 31 days</u>
<u>≥ 2**</u>	<u>≥ 6</u>	<u>At least once per 7 days</u>

01-13-LS1

~~* Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 20 and 100 tests and failures are determined on a per diesel generator basis. For the purpose of this schedule, only valid tests conducted after the completion of the preoperational test requirements of Regulatory Guide 1.108, Revision 1, August 1977, shall be included in the computation of the "Last 20 Valid Tests" and the "Last 100 Valid Tests." For the purpose of determining the required test frequency, the previous test failure count may be reduced to zero if the specific cause for the diesel unreliability has been identified and resolved; appropriate post maintenance operation and testing have been completed; and if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. These tests shall be in accordance with Specification 4.8.1.1.2a.2).~~

~~** The associated test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one.~~

TABLE 4.8-2a.

LOAD SEQUENCING TIMERS
ESF TIMERS

TIMER SETTINGS (s)

<u>COMPONENT</u>	<u>MINIMUM</u>	<u>NOMINAL</u>	<u>MAXIMUM</u>
1. <u>Bus F</u> Centrifugal Charging Pump No. 1	1.5	2	3
Safety Injection Pump No. 1	5	6	7
Containment Fan Cooler Unit No. 2	9	10	11
Containment Fan Cooler Unit No. 1	13	14	15
Component Cooling Water Pump No. 1	17	18	19.5
Auxiliary Saltwater Pump No. 1	20.8	22	23.2
Auxiliary Feedwater Pump No. 3	24.5	26	28
2. <u>Bus G</u> Centrifugal Charging Pump No. 2	1.5	2	3
Residual Heat Removal Pump No. 1	5	6	7.5
Containment Fan Cooler Unit No. 3	9	10	11.5
Containment Fan Cooler Unit No. 5	13	14	15
Component Cooling Water Pump No. 2	17	18	19
Auxiliary Saltwater Pump No. 2	20.8	22	23.2
Containment Spray Pump No. 1	24.5	26	28
3. <u>Bus H</u> Safety Injection Pump No. 2	1	2	3
Residual Heat Removal Pump No. 2	5	6	7
Containment Fan Cooler Unit No. 4	9	10	11
Component Cooling Water Pump No. 3	12.5	14	15
Auxiliary Feedwater Pump No. 2	17	18	19.5
Containment Spray Pump No. 2	20.8	22	23.2

01-20-LG

TABLE 4.8-2b.

LOAD SEQUENCING TIMERS
AUTO TRANSFER TIMERS

TIMER SETTINGS (s)

<u>COMPONENT</u>	<u>MINIMUM</u>	<u>NOMINAL</u>	<u>MAXIMUM</u>
1. <u>Bus F</u> Component Cooling Water Pump No. 1	4	5	6
Auxiliary Saltwater Pump No. 1	9	10	11
Auxiliary Feedwater Pump No. 3	13	14	15
Centrifugal Charging Pump No. 1	18.5	20	21.5
Containment Fan Cooler Unit No. 1	23.5	25	27
Containment Fan Cooler Unit No. 2	23.5	25	27
2. <u>Bus G</u> Component Cooling Water Pump No. 2	4	5	
—6			
Auxiliary Saltwater Pump No. 2	9	10	11
Centrifugal Charging Pump No. 2	18.5	20	21.5
Containment Fan Cooler Unit No. 3	23.5	25	27
Containment Fan Cooler Unit No. 5	23.5	25	27
3. <u>Bus H</u> Component Cooling Water Pump No. 3	4	5	6
Auxiliary Feedwater Pump No. 2	13	14	15
Containment Fan Cooler Unit No. 4	22	25	27

01-20-LG

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System required by LCO 3.8.2.2, and

01-41-M

b. One diesel generator capable of supplying the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.2.2, with-

1. A separate engine-mounted fuel tank containing a minimum volume of 250 gallons of fuel,

01-01-A

2. One supply train of the Diesel Fuel Oil Storage and Transfer system containing a minimum storage of 26,000 gallons* of fuel in addition to the fuel required for the other unit**.

01-31-A

(new) A lube oil inventory shall be within limits

01-48-M

(new) A starting air receiver pressure shall be within limits, and

01-48-M

(new) A turbocharger air assist air receiver pressure shall be within limits

01-64-M

APPLICABILITY: MODES 5 and 6 and during movement of irradiated fuel assemblies

01-42-M

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel or crane operations with loads over the fuel storage pool. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, and immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible, or with one required offsite circuit inoperable, declare affected required features with no offsite power available inoperable. With one de-energized Class 1E AC electrical power subsystem as a result of the inoperable circuit, enter applicable Actions of LCO 3.8.2.2.

01-44-LG

01-45-M

01-43-M

01-59-LS20

01-46-A

SURVEILLANCE REQUIREMENTS

NOTE:

The following SRs are not required to be performed: 4.8.1.1.2a.3), 4.8.1.1.2b.2) (auto-transfer timers), 4.8.1.1.2b.3), 4.8.1.1.2b.4), 4.8.1.1.2b.5), 4.8.1.1.2b.7)c), 4.8.1.1.2b.8), 4.8.1.1.2b.10), and 4.8.1.1.2e.

01-47-LS4

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the following requirements of applicable Specifications 4.8.1.1.1, 4.8.1.1.2, 4.8.1.1.3, and 4.8.1.1.4, except for Specifications 4.8.1.1.1.b.2) and 4.8.1.1.2.a.2)c), b.2) for ESF timers, b.6), b.7), b.10), and b.11) 4.8.1.1.1a) 4.8.1.1.2a.2), 4.8.1.1.2a.3), 4.8.1.1.2a.1), 4.8.1.1.2d, 4.8.1.1.3a.2), 4.8.1.1.2b.4), 4.8.1.1.2b.5), 4.8.1.1.2e, and 4.8.1.1.2b.2).

01-47-LS4

~~* The performance of Technical Specification Surveillance Requirement 4.8.1.1.3.c requires one fuel oil storage tank to be removed from service to be drained and cleaned. During this surveillance, the diesel generator fuel oil storage requirement for one unit operation in Modes 5 or 6 and one unit operation in Mode 6 with at least 23 feet of water above the reactor vessel flange or with the reactor vessel defueled is 35,000 gallons. The tank being cleaned may be inoperable for up to 10 days. For the duration of tank cleaning, temporary onsite fuel oil storage of 24,000 gallons will be maintained. Prior to removal of a tank from service, the offsite circuits required by Technical Specification 3.8.1.2.a will be verified to be OPERABLE.~~

01-01-A

~~** The performance of modifications to the diesel fuel oil storage and transfer system requires one fuel oil storage tank at a time to be drained and replaced with a new storage tank. During this period, the diesel generator fuel oil storage requirement for one or two unit operation in Modes 5 and 6 is 35,000 gallons. A total of up to 120 days may be required to complete the replacement of both tanks. For the duration of the tank replacement, temporary onsite storage of 30,000 gallons will be maintained. Prior to removal of a tank from service, the offsite circuits required by Technical Specification 3.8.1.2a. will be verified to be OPERABLE.~~

01-31-A

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following electrical busses shall be energized in the specified manner:

- a. ~~4160 volt Vital Bus F,~~
- b. ~~480 volt Vital Bus F,~~
- c. ~~4160 volt Vital Bus G,~~
- d. ~~480 volt Vital Bus G,~~
- e. ~~4160 volt Vital Bus H,~~ 03-01-LG
- f. ~~480 volt Vital Bus H,~~
- g. ~~120 volt Vital Instrument A.C. Bus 1 energized from its associated inverter connected to D.C. Bus 1,~~
- h. ~~120 volt Vital Instrument A.C. Bus 2 energized from its associated inverter connected to D.C. Bus 2,~~
- i. ~~120 volt Vital Instrument A.C. Bus 3 energized from its associated inverter connected to D.C. Bus 3,~~ 03-02-A
- j. ~~120 volt Vital Instrument A.C. Bus 4 energized from its associated inverter connected to D.C. Bus 2,~~
- k. ~~125 volt D.C. Bus 1 energized from Battery Bank 1, and its associated full capacity charger,~~
- l. ~~125 volt D.C. Bus 2 energized from Battery Bank 2, and its associated full capacity charger, and~~
- m. ~~125 volt D.C. Bus 3 energized from Battery Bank 3, and its associated full capacity charger.~~

The required AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

- a. ~~With one of the required 4160 volt and/or associated 480 volt vital busses not energized, re energize them~~ 03-01-LG
~~AC electrical power distribution subsystem inoperable, restore within 8 hours and within at least 16 hours from discovery of failure to meet the LCO, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~ 03-04-M

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

- b. ~~With one vital instrument A.C. bus inoperable restore to OPERABLE status not energized from its associated inverter, or with one inverter not connected to its associated D.C. bus, re energize the vital instrument A.C. bus from an alternate source within 2 hours and within at least 16 hours from discovery of failure to meet the LCO~~ or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; re-energize the vital instrument A.C. bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. 03-01-LG
03-04-M
- c. With more than one full capacity charger receiving power simultaneously from a single 480 volt vital bus or any D.C. bus not receiving power from its associated A.C. division, restore the system to a configuration wherein each charger is powered from its associated 480 volt vital bus within 14 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- d. ~~With one D.C. bus not energized from its associated battery bank and a full capacity charger, re energize it from its associated battery bank and a full capacity charger inoperable, restore to operable status within 2 hours and within at least 16 hours from discovery of failure to meet the LCO~~ or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. 03-01-LG
03-04-M
- ~~(new) With two Glass 1E Vital busses with inoperable distribution subsystems that result in a loss of safety function enter LCO 3.0.3 immediately~~ 03-05-A

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 ~~As a minimum, the following electrical busses shall be energized in the specified manner:-~~

- a. ~~One 4160 volt and its associated 480 volt A.C. vital bus,~~
- b. ~~Two 120 volt vital instrument A.C. busses energized from their associated inverters connected to their respective D.C. busses, and~~ 03-06-M
- c. ~~One 125 volt D.C. bus energized from its associated battery bank and full capacity charger supplied from its associated OPERABLE A.C. vital bus.~~

~~The necessary portion of the AC, DC, and AC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE~~

APPLICABILITY: ~~MODES 5 and 6 and during movement of irradiated fuel assemblies~~ 01-42-M

ACTION:

~~With any of the above required electrical busses inoperable, immediately declare associated supported required feature(s) inoperable, or not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible and immediately declare associated required residual heat removal subsystem(s) inoperable and not in operation.~~ 03-03-LS20
03-07-M

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

3/4.8.3 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 ~~The following Three Class 1E D.C. electrical sources shall be OPERABLE:~~

- ~~a. 125 volt D.C. Battery No. 1 and an associated full capacity charger,~~
- ~~b. 125 volt D.C. Battery No. 2 and an associated full capacity charger, and~~
- ~~c. 125 volt D.C. Battery No. 3 and an associated full capacity charger.~~

02-01-LG

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

~~With one of the required battery banks and/or full capacity chargers DC electrical power subsystems inoperable, restore the inoperable battery bank and/or full-capacity charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

02-01-LG

SURVEILLANCE REQUIREMENTS

4.8.3.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

- a. At least once per 7 ~~DI~~ days by verifying that:
 - 1) The parameters in Table 4.8-3 meet the Category A limits, and
 - 2) The total battery terminal voltage is greater than or equal to 130-volts on float charge.

02-06-LS22

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- b. ~~At least once per 92 days and within 7 days after a battery discharge with D.C. bus voltage below 118 volts, or battery overcharge with D.C. bus voltage above 145 volts, by:~~ 02-17-LS2
- 1) ~~Verifying that the parameters in Table 4.8-3 meet the Category B limits, and within 7 days after a battery discharge with D.C. bus voltage below 118 volts, or battery overcharge with D.C. bus voltage above 145 volts.~~ 02-24-LG
02-23-A
 - 2) ~~Verifying there is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than or equal to 150×10^{-6} ohm,* and~~
 - 3) ~~Verifying that the average electrolyte temperature of 10 of the connected representative cells is above greater than or equal to 60°F.~~ 02-18-LG
02-22-A
- c. At least once per 18 months by verifying that:
- 1) ~~The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.~~ 02-02-A
 - 2) ~~The Remove visible terminal corrosion, verify battery cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material.~~ 02-03-M
 - 3) ~~The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohm,* and~~ 02-24-LG
 - 4) ~~The battery charger will supply at least 400 amperes at 130 volts for at least 4 hours~~
- d. ~~At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual and/or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test. This surveillance shall not be performed in MODES 1, 2, 3, or 4 but credit may be taken for unplanned events that satisfy this SR.~~ 02-04-M
- e. ~~At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test. This performance discharge test or modified performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.3.1d. This surveillance shall not be performed in MODES 1, 2, 3, or 4 but credit may be taken for unplanned events that satisfy this SR, and~~ 02-05-M
02-04-M

- f. At least once per 18 months during shutdown, by giving performance discharge tests, or a modified performance discharge test, of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating, and at least once per 24 months when battery has reached 85% of service life for the application with capacity $\geq 100\%$ of manufacturer's rating, this surveillance shall not be performed in Mode 1, 2, 3, or 4, but credit may be taken for unplanned events that satisfy this SR.

02-21-LS1

02-05-M

02-18-LG

02-04-M

~~*The resistance of cell to cell connecting cables does not have to be included.~~

02-24-LG

TABLE 4.8-3

BATTERY SURVEILLANCE REQUIREMENTS

PARAMETER	CATEGORY A(1)	CATEGORY B(2) CATEGORY B CATEGORY C	<u>02-14-A</u>
	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE(3) VALUE FOR EACH CONNECTED CELL
Electrolyte Level <u>02-07-A</u>	>Minimum level indication mark, and $\leq \frac{1}{4}$ " above maximum level indication mark	>Minimum level indication mark, and $\leq \frac{1}{4}$ " above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	> 2.13 volt (6) <u>02-12-LG</u>	> 2.07 volts
Specific Gravity(4)	$\geq 1.195(5)$	≥ 1.190	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.200	Average of all connected cells $\geq 1.190(5)$

- (1) For any Category A or B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that pilot cell level and float voltage meet Cat. B allowance value within 1 hour, within 24 hours and once every 7 days all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 ~~30~~ days. 02-08-LS21
02-09-M
- ~~(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days. 02-08-LS21~~
- (3) Any Category B parameter not within its allowable value indicates an inoperable battery.
- (4) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging is < 2 amps when on float charge. 02-10-A
- (5) Or battery charging current is less than 2 amps when on charge. This is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance. 02-11-A
- ~~(6) Corrected for average electrolyte temperature. 02-12-LG~~
- (new)(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing. 02-07-A

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 ~~As a minimum, one 125-volt battery bank and an associated full capacity charger shall be OPERABLE. DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem required by LCO 3.8.2.2.~~

02-01-LG

02-13-A

APPLICABILITY: MODES 5 and 6 and during movement of irradiated fuel assemblies.

01-42-M

ACTION:

~~With the required battery bank and/or full capacity charger one or more required DC electrical power subsystems inoperable. Declare affected required feature(s) inoperable; or immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required battery bank and/or full capacity charger DC electrical power subsystems to OPERABLE status as soon as possible.~~

02-19-LS20

02-01-LG

SURVEILLANCE REQUIREMENTS

NOTE

~~The following SRs are not required to be performed: SR 4.8.3.1d, SR 4.8.3.1e and SR 4.8.3.1f.~~

02-15-LS4

4.8.3.2 ~~The above required 125-volt battery bank and charger DC Sources shall be demonstrated OPERABLE in accordance with Specification 4.8.3.1.~~

02-01-LG

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

04-01-R

LIMITING CONDITION FOR OPERATION

~~3.8.4.1 The thermal overload protection and bypass devices, integral with the motor starter, of each valve[#] used in safety systems shall be OPERABLE.~~

~~APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.~~

ACTION:

~~With one or more of the thermal overload protection and/or bypass devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected valves.~~

SURVEILLANCE REQUIREMENTS

~~4.8.4.1 The above required thermal overload protection and bypass devices shall be demonstrated OPERABLE:~~

~~a. At least once per 18 months, by the performance of a TRIP ACTUATION DEVICE OPERATIONAL TEST of the bypass circuitry for those thermal overload devices which are either:~~

- ~~1) Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or~~
- ~~2) Normally in force during plant operation and bypassed under accident conditions.~~

~~b. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of:~~

- ~~1) All thermal overload devices which are not bypassed, such that each non bypassed device is calibrated at least once per 6 years, and~~
- ~~2) All thermal overload devices which are continuously bypassed, such that each continuously bypassed device is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor operator when the thermal overload device is OPERABLE and not bypassed at least once per 6 years.~~

~~# See AD13.DC1 for List of Motor Operated Valves Thermal Overload Protection and Bypass Devices~~

ELECTRICAL POWER SYSTEMS

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

04-01-R

LIMITING CONDITION FOR OPERATION

~~3.8.4.2 For each containment penetration provided with a containment penetration overcurrent protective device(s), each device shall be OPERABLE.~~

~~APPLICABILITY: MODES 1, 2, 3, and 4.~~

ACTION:

~~With one or more of the above required containment penetration conductor overcurrent protective device(s) inoperable:~~

- ~~a. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated protective device or racking out or removing the inoperable protective device within 72 hours, declare the affected system or component inoperable, and verify the associated protective device to be tripped or removed, or the inoperable protective device racked out or removed at least once per 7 days thereafter; or~~
- ~~b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

SURVEILLANCE REQUIREMENTS

~~4.8.4.2 Protective devices required to be operable as containment penetration overcurrent protective devices shall be demonstrated OPERABLE:~~

- ~~a. At least once per 18 months:~~
 - ~~1) By verifying that the medium voltage 12 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers and performing the following:~~
 - ~~a) A CHANNEL CALIBRATION of the associated protective relays,~~
 - ~~b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breaker and overcurrent control circuit function as designed, and~~

SURVEILLANCE REQUIREMENTS (Continued)

- ~~c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.~~
- ~~2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of "drawout" type circuit breakers shall consist of a CHANNEL CALIBRATION of the associated solid state trip device for both the long time delay trip element and the short time delay element along with a breaker functional test. Testing of molded case circuit breakers shall consist of injecting a current with a value equal to 200% (for D.C. breakers) and 300% (for A.C. breakers) of the pickup of the time delay element and verifying that the circuit breaker operates within the time delay band for that current specified by the manufacturer. The instantaneous element of molded case circuit breakers shall be tested by injecting a current equal to 25% +40% of the pickup value of the element and verifying that the circuit breaker trips with no intentional time delay. Circuit breakers found out of tolerance during functional testing shall be replaced prior to resuming operation. Circuit breakers that fail to trip magnetically before the withstand capability of the penetration conductor is reached shall be declared inoperable. Circuit breakers that fail to trip thermally before the manufacturer's maximum tolerance shall be declared inoperable. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested; and~~
- ~~3) By verifying that the thermal overload devices integral with the motor starters, used for penetration overcurrent protection, are OPERABLE by selecting a representative sample of at least 10% of the motor overload devices and performing a CHANNEL CALIBRATION. Motor overloads found inoperable shall be restored to OPERABLE status prior to resuming operation. For each motor overload device found inoperable, a CHANNEL CALIBRATION shall be performed on an additional representative sample of at least 10% of all the motor overload devices of the inoperable type until no more failures are found or a CHANNEL CALIBRATION has been performed on all motor overload devices of that type.~~
- ~~b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.~~

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. **Deletions** - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. **Additions** - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. **Modifications** - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. **Administrative** - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions** - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions** - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications** - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative** - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

Methodology For Mark-Up of Current TS
(Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers (1 Pages)
Description of Changes (14 Pages)

TECHNICAL SPECIFICATION CHANGE

SECTION 3/4.8

TECHNICAL SPECIFICATION TITLE	CHG. NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON .
A.C. Sources - Operating	01	3.8.1.1	3.8.1.1	3.8.1.1	3.8.1.1
A.C. Sources - Operating	01	3.8.1.2	3.8.1.2	3.8.1.2	3.8.1.2
D.C. Sources - Operating	02	3.8.2.1	3.8.2.1	3.8.2.1	3.8.3.1
D.C. Sources - Shutdown	02	3.8.2.2	3.8.2.2	3.8.2.2	3.8.3.2
Onsite Power Distribution - Operating	03	3.8.3.1	3.8.3.1	3.8.3.1	3.8.2.1
Onsite Power Distribution - Shutdown	03	3.8.3.2	3.8.3.2	3.8.3.2	3.8.2.2
Electrical Equipment Protective Devices - Motor-Operated Valves Thermal Overload Protection Devices	04	N/A	N/A	N/A	3.8.4.1
Equipment Protective Devices - Containment Penetration Conductor Overcurrent Protective Devices	04	N/A	N/A	3.8.4	3.8.4.2

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	A	<p>TS requirement for minimum diesel generator (DG) day tank fuel oil volume and fuel transfer pump would be moved to improved Technical Specification (ITS) Surveillance Requirement (SR) 3.8.1.4 and SR 3.8.1.6. This change does not result in a change to any technical requirement and is administrative in nature. This change is consistent with NUREG-1431.</p> <p>TS requirements for diesel fuel oil (DFO) storage would be moved to ITS LCO 3.8.3.</p>
01-02	A	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-03	LS5	<p>CTS 3.8.1.1 ACTION requirements to demonstrate DG OPERABILITY is deleted. ACTIONS currently require all OPERABLE DGs that have not been tested within the previous 24 hours to be started within 24 hours to demonstrate their OPERABILITY in the event that one or both offsite A.C. circuits become inoperable. The normal TS surveillance testing already demonstrates that an OPERABLE DG is capable of performing its safety function. The inoperability of an offsite circuit does not in any way effect the DG(s). Therefore, this testing is unnecessary. This change is consistent with Generic Letter (GL) 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirement for Testing During Power Operation," and with NUREG-1431.</p>
01-04	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-05	LS6	<p>CTS 3.8.1.1 ACTIONS require that the remaining OPERABLE DG(s) be started to demonstrate OPERABILITY [within 24 hours] in the event a DG becomes inoperable due to any cause other than preventive maintenance or testing. The intent of this additional testing is to determine if a common cause failure exists and to provide added assurance that the remaining OPERABLE DG(s) is capable of supplying emergency power. The CTS also requires that this testing be completed even if the inoperable DG is restored to OPERABLE. However, this requirement results in unnecessary testing when there is no common cause failure. This proposed change incorporates the NUREG-1431 provision of "determining the OPERABLE DG(s) are not inoperable due to a common cause failure" and deletes the requirement to complete the test even if the ACTION is exited, which avoids unnecessary testing of OPERABLE DG(s), into ACTION requirements. The proposed change is acceptable because it would continue to assure that a common cause failure does not exist while reducing wear on the DG(s) as endorsed by GL 84-15, "Proposed staff actions to improve and maintain diesel generator reliability," and GL 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirement for Testing During Power Operation."</p>

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
		This change is consistent with NUREG-1431.
01-06	LS7	<p>CTS 3.8.1.1 ACTION requirements allowed time for verification of required redundant feature OPERABILITY would be increased from 2 to 4 hours. The shutdown requirement would be replaced with a requirement to enter the required feature's ACTION statement.</p> <p>The ACTION currently requires that with one DG inoperable, all required safety equipment that depends on the remaining operable DG(s) be verified OPERABLE. If these requirements are not met within 2 hours, a unit shutdown is required. The proposed changes to the ACTION requirement would increase the allowed time from 2 hours to 4 hours. In addition, the Safety Function Determination Program would require an immediate evaluation of redundant features for inoperability. This would result in entering the required feature's TS ACTION statement.</p> <p>This change is acceptable because it would assure that appropriate ACTIONS are taken to assure the OPERABILITY of required systems before requiring a plant shutdown and is consistent with NUREG-1431.</p>
01-07	LS3	<p>CTS SR to start all DGs simultaneously at least once per 10 years or after any modification which could affect DG starting interdependence would be revised to delete the requirement to perform the SR during shutdown. This test does not require the DG to be paralleled and loaded onto the offsite source. In addition, the DG remains OPERABLE during this SR. This change is consistent with NUREG-1431.</p>
01-08	LS8	<p>CTS 3.8.1.1 ACTION requirement to demonstrate DG OPERABILITY within 8 hours would be increased to 24 hours. Extending the time required to demonstrate the OPERABILITY of the remaining OPERABLE DGs would allow 24 hours to determine the OPERABLE DGs are not inoperable due to a common cause failure or demonstrate OPERABILITY of the remaining OPERABLE DGs by testing. This proposed change avoids unnecessary DG testing. This change would allow the same time period (24 hours) for determining OPERABILITY of the remaining OPERABLE DG(s) as the ACTION for one inoperable DG currently allows and is consistent with the recommendations of GL 84-15, "Proposed staff actions to improve and maintain diesel generator reliability." This proposed change is consistent with NUREG-1431.</p>
01-09		Not used.
01-10	M	<p>CTS 3.8.1.1 ACTIONS would be revised to require transition to MODE 5 within 30 hours following transition to MODE 3 upon failure to meet the requirements of these ACTION statements. This change would require placing the plant in a MODE in which the limiting condition for operation (LCO) no longer applies. This change is consistent with NUREG-1431.</p>
01-11	TR1	<p>Several CTS SRs would be revised to clarify that an actual actuation signal, as well as a test or simulated signal, may be used. This change would allow satisfactory automatic systems actuations from actual signals to be used to fulfill the requirements of the SR. This proposed change is acceptable because data obtained during the performance of the surveillance must still be obtained and the acceptance criteria must be satisfied before credit can be taken for the actual actuation, e.g., an unplanned event.</p>

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-12	A	The footnote allowing an engine prelube period would be applicable to all DG SRs involving an engine start. The DGs are maintained in a prelubed condition, and therefore this change in format does not add or reduce any requirement. This change is administrative in nature and consistent with Regulatory Guide (RG) 1.9, "Selection and Diesel Generator Set Capacity for Standby Power Supplies," Rev. 3 and with NUREG-1431.
01-13	LS1	The CTS requiring DG reports and DG test schedule tables would be deleted, and other CTS would be revised to reflect deletion of the reporting requirement and DG test schedule. These proposed changes are consistent with the recommendations of GL 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," dated May 31, 1994. With the proposed changes, accelerated testing of the DGs and DG special reporting requirements would be deleted.
01-14	A	CTS requirement of "standby status" would be replaced with "ready-to-load operation." This terminology is more precise. This change is consistent with NUREG-1431.
01-15	A	The requirement to start the DG from "ambient condition" would be replaced with "standby conditions," and description of these conditions added to the Bases. The phrase "standby conditions" more accurately reflects DG testing conditions. The DCPD DGs are provided with jacket water and lube oil warming systems. Also, the engine bearings are lubricated and ready for operation via the lube oil circulating pumps, which run continuously until the DG is started. If the lube oil warming system should become inoperable, the lube oil temperature is monitored to ensure it remains above the manufacturer's recommended temperature. This is consistent with [License Amendment Request (LAR) 97-02 (submitted 2/27/97) and] the definition provided in RG 1.9, "Selection and Diesel Generator Set Capacity for Standby Power Supplies," Rev. 3 of "standby conditions" and does not result in any technical relaxations or additions to the current operating understanding of "ambient conditions" in the CTS. This change is consistent with NUREG-1431.
01-16	LG	The normal monthly DG start SR would be revised to delete the method by which the DG is started. Currently, the TS specifies that the DG shall be started by using one of the start signals. Deleting the listing of the start signals that may be used to start the DG would eliminate unnecessary information contained in the TS. The specific method for starting a DG would be moved to the Bases. This change is consistent with NUREG-1431.
01-17	LS10	The SR to rapidly load the DG in less than or equal to 60 seconds would be deleted. Rapidly loading the DG introduces unnecessary mechanical stresses and engine wear. This change is consistent with GL 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirement for Testing During Power Operation," and NUREG-1431.
01-18	LS11	The current DG loading requirements are specified in SRs as "greater than or equal to" These SRs would be revised to specify the DG is loaded between a range of kW values. This change is consistent with NUREG-1431.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-19	LS12	A footnote would be added to several SRs stating that momentary transients outside the load and/or power factor range do not invalidate the test, since DG loading could change during this test due to changing bus conditions. Some load fluctuation is expected and would occur during DG operation under post-accident conditions. Therefore, this change is acceptable and should not invalidate this test. This change is consistent with NUREG-1431.
01-20	LG	Several SRs would be moved to licensee-controlled documents. Moving these requirements would allow changes to these requirements without prior Commission approval and will further the Commission's goal of TS improvements, as delineated in NRC policy statements, without reducing the level of plant safety. The SRs that are being moved reflect normal design, maintenance or line-up activities/descriptions rather than features specifically needed to successfully mitigate a design basis accident (DBA) or design transient. These proposed changes are consistent with NUREG-1431.
01-21	M	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-22	M	An additional requirement would be added to TS 3.8.1.1 ACTION for two offsite circuits inoperable, requiring that concurrent with inoperability of required redundant feature(s), the required feature also be declared inoperable. This requirement is intended to provide assurance that an event with a coincident single failure would not result in a complete loss of redundant, required safety functions. The addition of this requirement is consistent with NUREG-1431.
01-23	LG	The DG partial load rejection SR would be revised to replace the requirement that the load be \geq [508] kW with "a load greater than or equal to its single largest post-accident load." The actual load and its corresponding horsepower and kW ratings would be included in the TS Bases. This change would eliminate unnecessary information contained in the TS and potentially reduce the need for a future LAR. This proposed change is consistent with NUREG-1431.
01-24	LS13	<p>The DG partial load rejection SR requirement that voltage be maintained would be replaced with a requirement that within 2.4 seconds following load rejection, the voltage is within steady state requirements. The voltage specified is consistent with the design range of the equipment powered by the DG.</p> <p>The 2.4 second allowance for voltage and frequency is based on the recommendations of RG 1.9, "Selection and Diesel Generator Set Capacity for Standby Power Supplies," Rev. 3, for recovery from transients caused by the disconnection of the largest single load. RG 1.9, Rev. 3, states that voltage and frequency should recover to within 10 percent and 2 percent of nominal, respectively, within 60 percent of the load sequence interval. The load sequence interval following addition of the largest load is 4 seconds. This change is consistent with NUREG-1431.</p>

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

CHANGE NUMBER	NSHC	DESCRIPTION
01-25	M	<p>The DG partial load rejection SR requirement to maintain frequency at 60 ± 3 Hz would be revised to require frequency be maintained at ≤ 63 Hz during load rejection, and an additional requirement would be added that the frequency recover to 60 ± 1.2 Hz within 2.4 seconds following load rejection. This more restrictive frequency recovery specification is consistent with RG 1.9, "Selection and Diesel Generator Set Capacity for Standby Power Supplies," Rev. 3, and the design range of the equipment powered by the DG. This change is consistent with NUREG-1431.</p>
01-26	M	<p>In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, power factor requirements would be added to several SRs.</p> <p>A power factor requirement of less than or equal to 0.9 lagging would be added to the DG partial load rejection SR as a note. This note would apply when the SR is performed with the DG synchronized with offsite power. A power factor requirement would be added to the DG full load rejection and 24-hour load run SRs. The addition of a power factor requirement is consistent with NUREG-1431.</p> <p>For DCCP, the single largest DG load is a centrifugal charging pump (CCP). This power factor is based on the actual design basis inductive loading demand of a CCP assumed in PG&E design calculations.</p> <p>A power factor requirement of less than or equal to 0.87 lagging would be added to the DG full load rejection and 24-hour load run SRs. This power factor is representative of the actual design basis inductive loading that the DGs would experience, and based on the load demands listed in PG&E design calculations (Final Safety Analysis Report (FSAR) Update Table 8.3-5 loads will be revised in the next FSAR Update to include these updated loads). The power factors corresponding to the inductive loading listed for the 4160 volt vital buses are as follows: 0.872, 0.870, and 0.886 for buses F, G, and H, respectively.</p> <p>Also, the DG 24-hour load run SR would be revised to add a note stating that momentary transients outside the power factor range do not invalidate the test, since power factor could change during this test due to changing bus conditions. Similar to real power loading, power factor fluctuation is expected and should not invalidate this test. PG&E believes the practice of monitoring and recording power factor every 15 minutes during the overload part of the 24 hour load test and once every hour for the remaining 22 hours is sufficient to ensure the DG power factor is within its range. DG power factor found out of the range and immediately returned to less than or equal to 0.87 would not invalidate the TS 24-hour load run SR.</p>
01-27	LS9	<p>The DG full load rejection SR maximum DG voltage following load rejection would be increased from 4580 V to 6200 V.</p> <p>The CTS voltage requirement of less than 4580 volts, which is based on the vital bus nominal voltage of 4160 volts plus 10 percent, is too restrictive, and not representative of potential SR testing overshoot voltage values. The proposed limit will allow flexibility in the adjustment of the DG droop settings to provide stable DG operation with the power factor requirements of ITS SR 3.8.1.10 while the DG is paralleled to offsite power for surveillance testing. This change is consistent RG 1.108, "Periodic Testing of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Rev. 1, and with NUREG-1431.</p>

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

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NSHC

DESCRIPTION

01-28	LG	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-29	A	<p>TS 4.8.1.1.2b.7) requires that at least every 18 months, during shutdown, each DG be demonstrated OPERABLE by simulating a loss of offsite power (LOOP) in conjunction with a safety injection (SI) test signal. Included in TS 4.8.1.1.2b.7) is part c), which requires verification that all automatic DG trips, except engine overspeed, low lube oil pressure, and generator differential, are bypassed when the DG trip cutout switch is in the cutout position and the DG is aligned for automatic operation.</p> <p>Including TS 4.8.1.1.2b.7)c) with the other tests which simulate proper DG response to a LOOP in conjunction with an SI test signal implies that the automatic DG trips, except as noted above, are bypassed only on a LOOP in conjunction with an SI signal. However, the design of the DCP DGs is such that all DG trip functions, with the exceptions noted above, are bypassed when the DG is started automatically (on loss of standby power, or SI signal, or both). Therefore, to clarify the DCP TS, TS 4.8.1.1.2b.7)c) would be separated from the requirements of TS 4.8.1.1.2b.7) and included as new TS 3.8.1.13. This change reorganizes these requirements but does not add or remove any technical requirements; therefore, this change is administrative in nature. This is consistent with NUREG-1431.</p>
01-30	LG	<p>The DG 24-hour full load SR voltage and frequency start requirements would be moved to the Bases. The current SR requires verification of DG voltage and frequency after start signal for the 24-hour load run. However, the purpose of the 24-hour load test is to verify the DG can operate, fully loaded, for an extended period of time. The verification of voltage and frequency following the start signal does not contribute to verifying the long term load carrying capability of the DG. This proposed change is consistent with NUREG-1431.</p>
01-31	A	<p>Cycle-specific information that is no longer applicable at the time the ITS is approved by the NRC would be deleted from the TS. This administrative change clarifies the TS by removing unnecessary information. This change is consistent with NUREG-1431.</p>
01-32	M	<p>This proposed change would revise TS 3.8.1.1 to require that with one or more A.C. electrical power sources inoperable, all A.C. electrical power sources be restored to OPERABLE status within 10 days from the time the first A.C. electrical power source became inoperable. The 10-day limit is based upon the addition of the time limit required for restoration of one offsite electrical source (72 hours) and one DG (7 days). This proposed change would add a more restrictive requirement to limit total time for not meeting the ITS LCO. This will provide additional assurance of the availability of the A.C. sources. The addition of a requirement limiting overall time for not meeting the LCO is consistent with NUREG-1431.</p>
01-33	LS15	<p>The SR to check for and remove accumulated water from the DG day tanks after each operation of the DG for greater than one hour would be deleted. Based upon operating experience, checking for accumulated water more often than once per 31 days is unwarranted and unnecessarily frequent. Deleting this requirement would eliminate unnecessary information contained in the TS. This change is consistent with NUREG-1431.</p>
01-34		Not used.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

CHANGE NUMBER

NSHC

DESCRIPTION

01-35	M	Adding the requirement to the 10-year simultaneous DG start SR to verify that the DGs achieve steady state voltage and frequency within the required time demonstrates that proper voltage and frequency performance is attained. This change is consistent with NUREG-1431.
01-36	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
01-37	LG	<p>The specific SRs for sampling new DFO prior to addition to the storage tanks and for sampling the stored DFO once every 31 days would be moved to the Diesel Fuel Oil Testing Program, which would be referenced by SR 3.8.3.3 and described in TS 5.5.13. The Diesel Fuel Oil Testing Program will function similar to other programs currently described in the Administrative Controls section of the TS.</p> <p>SR 3.8.3.3 would require that DFO properties of new and stored DFO are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program. In addition, the SR 3.8.3 Bases will contain additional descriptive material regarding the DFO testing requirements of the current SRs.</p> <p>These changes are consistent with NUREG-1431.</p>
01-38	LG	The fuel oil storage tank cleaning SR would be moved to licensee-controlled documents. This change is consistent with NUREG-1431.
01-39	LG	The fuel oil transfer piping leak test SR would be moved to licensee controlled documents. This change is consistent with NUREG-1431.
01-40	LS25	The requirements for staggered DG testing would be deleted and replaced with a requirement to test each DG every 31 days. Deleting staggered DG testing is consistent with NUREG-1431 and results in no change in the Frequency of testing for each DG. This change is acceptable because studies have shown that staggered testing has negligible impact on equipment reliability. It is not as safety significant as originally thought, it is difficult to implement operationally, and may increase the chance of human error test performance.
01-41	M	The AC Sources - Shutdown LCO would be changed to specify that the required AC sources must be powering, or capable of powering, the required AC distribution busses. This change is an additional restriction consistent with the implicit assumptions for operation during shutdown conditions. This change is consistent with NUREG-1431. (See also CN 03-06-LS26)
01-42	M	CTS 3.8.1.2, CTS 3.8.2.2, and CTS 3.8.3.2 Applicability would be revised to add, "during movement of irradiated fuel assemblies." Addition of this restriction ensures that systems necessary to mitigate a fuel handing accident are available when the unit is not operating in MODEs 1-6. This change is consistent with NUREG-1431.
01-43	M	The AC Sources - Shutdown ACTION statement would be revised to specify that the offsite circuit must be capable of supplying power to the required onsite buses. This assures the single operable circuit is performing a vital function. Explicitly requiring entry into the applicable ACTIONS of LCO 3.8.10 clarifies the appropriate ACTION to take if the onsite distribution train becomes deenergized. This is consistent with NUREG-1431.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-44	LG	The AC Sources - Shutdown ACTION statement reference to crane operations with loads over the spent fuel pool would be moved to licensee-controlled documents. This is consistent with NUREG-1431 and is an example of moving details not required for operational safety out of the TS.
01-45	M	The AC Sources - Shutdown ACTION statement requirement to initiate corrective action is expanded to match the Applicability of the TS (MODE 5 and 6). This will replace the current requirement to "immediately initiate corrective action" which is applicable only if reactor coolant system (RCS) loops not full in MODE 5 and less than 23 feet of water above the reactor vessel flange in MODE 6. This more restrictive requirement continues to ensure that qualified power is provided to systems required operable in these MODES. This change is consistent with NUREG-1431.
01-46	A	A Note would be added to the AC Sources - Shutdown ACTION statement to enter the Distribution Systems LCO if one required electrical distribution subsystem has no electrical power. The change is already implicitly a part of the CTS requiring one offsite circuit and the onsite distribution system required to support Onsite Power Distribution-Shutdown LCO. The added note clarifies that the need is to support necessary equipment. This change does not add or remove any technical requirements and is therefore administrative in nature. This change is consistent with NUREG-1431.
01-47	LS4	The SRs required for AC sources OPERABILITY in MODES 5 and 6 would be revised to include only those SRs which are applicable. SRs that are not applicable are those that depend on engineered safety feature (ESF) actuation signals (which are not required to be operational during MODE 5 and 6) and automatic load sequencing (most of these loads are not required in MODE 5 and 6). The 10-year simultaneous auto-start of all DGs is also not applicable to MODE 5 and 6. The note listing exceptions to SR required for MODES 5 and 6 in CTS 4.8.1.2 would be revised to include additional SRs. SRs that are applicable but not required to be performed are those that place a DG in parallel with offsite power which increases the probability of a station blackout. This change is consistent with NUREG-1431 and is acceptable because it would assure the performance of SRs that are necessary and safe to perform for the plant Conditions.
01-48	M	<p>New ITS LCO 3.8.3 would be added. This LCO adds new TS requirements for DG lube oil and starting air. The Applicability of this LCO is tied to DG OPERABILITY requirements.</p> <p>This provides conditions for these DG support systems which allow additional time before requiring that the DG be declared inoperable. However, this is considered a more restrictive generic change since the CTS did not previously include DG lube oil and starting air requirements.</p> <p>The Conditions provided for these support systems address the situation where the support system is degraded but still capable of supporting the associated DG. The time allowed by the new TS Conditions to correct the support system problem before declaring the DG inoperable is acceptable based on the remaining capacity or capability of the systems and the low probability of an event occurring during the allowed time that would require the support systems' full capacity/capability.</p>

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

CHANGE NUMBER

NSHC

DESCRIPTION

The new requirements for lube oil inventory (ITS Condition B. and ITS SR 3.8.3.2) verify sufficient lube oil is available to support DG operation. The lube oil storage volume requirement is based upon a percentage of the consumption rate of DFO and the storage assignment is therefore based upon the DBA calculation for diesel fuel oil usage.

The new requirements for air start receiver pressures (ITS Condition E. and ITS SR 3.8.3.4) verify sufficient capacity of the air start receivers.

These changes are consistent with NUREG-1431.

01-49

LS16

NUREG-1431 LCO 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," includes Conditions and Required ACTIONS that allow reduced DFO inventory for up to 48 hours before requiring that the associated DG be declared inoperable. A new Condition would address stored DFO with total particulates out of limit and allow 7 days for restoration. A new Condition would address new fuel oil with properties not within limits and allow 30 days for restoration. The additional allowed outage time (AOT) for DFO is acceptable based on the remaining capacity of the DFO system and the low probability of an event occurring during the time that the DFO requirements were not met. This change is consistent with NUREG-1431.

01-50

LS17

CTS 3.8.1.1 ACTION shutdown requirement would be replaced with a requirement to enter the required feature's ACTION statement. The ACTION currently requires that with one DG inoperable, all required safety equipment that depends on the remaining OPERABLE DG(s) be verified OPERABLE. If these requirements are not met, a unit shutdown is required. Rather than requiring a unit shutdown, the proposed change would require declaring inoperable the required safety equipment powered from an inoperable DG. This would result in entering the required feature's TS ACTION statement. These ACTIONS would assure that specific measures, appropriate for a loss of safety function associated with inoperable redundant features, would be taken. From a safety standpoint it is preferable to avoid an unnecessary plant shutdowns. This change is consistent with NUREG-1431 and is acceptable because it would continue to assure the OPERABILITY of required features without requiring an unnecessary plant shutdown.

01-51

LS18

Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

01-52

M

Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

01-53

M

A Note would be added to the SR for monthly DG load test to allow testing on only one DG at a time. Consistent with current practice, this prevents having two DG synchronized and loaded on offsite power at one time. This reduces the risk of "common cause" failure. This change is consistent with NUREG-1431.

01-54

LS19

Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

01-55

M

A Note would be added prohibiting several SRs from being performed in MODES 1 through 4. This change is consistent with NUREG-1431.

01-56

M

Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-57	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-58	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-59	LS20	AC Sources - Shutdown ACTION statement would be revised to provide the alternative to "declare affected required features with no offsite power available inoperable." Since the circuit OPERABILITY requirements are in terms of powering all distribution systems required to be OPERABLE by LCO 3.8.10, the Required ACTION for one circuit inoperable would be revised to add, "declare affected required features with no offsite power available inoperable," as an alternative to the other Required ACTIONS. In the event one or more required load centers, motor control centers, buses, etc., are not capable of being powered via an offsite circuit, it may not be necessary to suspend all CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel. In this case, conservative ACTION can be assured if all required equipment without qualified offsite power availability is declared inoperable and the associated ACTIONS taken. This change is consistent with NUREG-1431.
01-60	LS24	Consistent with NUREG-1431; the surveillance interval for verifying that other properties are within limits for ASTM 2D fuel oil is changed from "within 30 days" to "within 31 days" after obtaining a sample. The fuel properties that can have an immediate detrimental impact on diesel combustion, (i.e., API gravity, kinematic viscosity, flash point and appearance) are verified prior to addition to the storage tank. The "other properties" may be analyzed after addition to the tank. The 31 day verification interval for these properties is acceptable because the fuel properties of interest, even if they are not within their stated limits, would not have an immediate affect on DG operation. The CTS 30 day verification interval was probably chosen because it was a convenient time interval for sending the sample and receiving the results from the laboratory selected for testing. NUREG-1431 has selected a 31 day testing interval. The 1 day increase in the interval would not have a significant affect on the acceptability of the DFO.
01-61	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-62	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-63	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-64	M	<p>The new requirements for turbocharger air assist air receiver pressures (ITS 3.8.3 LCO, Condition F. and ITS SR 3.8.3.6) verify sufficient capacity of the turbocharger air assist air receivers.</p> <p>This provides Conditions for these DG support systems which allow additional time before requiring that the DG be declared inoperable. However, this is considered a more restrictive generic change since CTS did not previously include turbocharger air assist air requirements.</p> <p>This change is consistent with the intent of NUREG-1431.</p>

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-65	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-66	TR3	<p>This SR currently requires that at least once per 10 years or after any modifications which could affect emergency diesel generator (EDG) interdependence, during shutdown, and verify that both EDGs accelerate to [at least 514 RPM] in less than or equal to [12 seconds]. It is being proposed that this SR be revised to eliminate the requirement to perform the test after any modifications which could affect EDG interdependence.</p> <p>This SR can be considered to be the "redundant unit test" in accordance with RG 1.9, "Selection and Diesel Generator Set Capacity for Standby Power Supplies," Rev. 3. This test demonstrates that by starting and running both redundant units simultaneously, potential "common cause failure" that may be undetected in single EDG unit tests do not occur. The proposed change to this SR will make it consistent with the ITS SR 3.8.1.20. The elimination of the requirement to perform this SR after any modification which could affect EDG interdependence is justified based upon the ability of the modification process to detect concerns related to the interdependence of the EDGs.</p>
02-01	LG	The list of batteries and chargers in the CTS DC Sources - Operating LCO and ACTION requirement would be moved to the Bases. This deletes descriptive information from the TS, consistent with NUREG-1431.
02-02	A	The phrase, "that could degrade battery performance," would be added to clarify the purpose of the battery inspection SR consistent with TSTF-38. This change does not add or remove any technical requirements and is administrative in nature.
02-03	M	The requirement to remove visible terminal corrosion would be added to the SR verifying on an 18-month frequency that cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material. These elements of a visual inspection are consistent with IEEE-450, 1995. This change is consistent with NUREG-1431.
02-04	M	A Note would be added to SRs that these SRs are not to be performed in MODEs 1, 2, 3, or 4. Since these surveillances discharge the battery such that it would not have capacity left to perform its required function, this SR must be performed when the battery is not required to support an operable vital bus. The addition of this Note is consistent with NUREG-1431.
02-05	M	The SR would be changed to allow performing a modified performance discharge test instead of the performance discharge test. The modified performance test is a more severe test and envelopes the battery service discharge test. The results of the modified performance test provides assurance of the battery capability as well as battery capacity. This change is consistent IEEE-450, 1995 and with NUREG-1431.
02-06	LS22	Consistent with industry Traveler TSTF-115, this change would allow the extension of the surveillance frequency verification for battery terminal voltage while on float charge, and for Category A battery cell parameters from 7 days to 31 days in accordance with the recommended frequency of at "least monthly" identified in IEEE 450-1995, Section 4.3.1.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
02-07	A	A Note would be added to the battery SRs table for Category A and B limits. This Note would allow the electrolyte level to temporarily increase above the maximum level during equalizing charges provided it does not overflow. This electrolyte level excursion is due to gas generation during the charge and is expected to return to normal after the charge. Therefore, the level excursion is acceptable. This change is consistent with NUREG-1431.
02-08	LS21	Notes contained in current battery SRs table for Category A and B limits would be combined in new LCO 3.8.6, "Battery Cell Parameters," as Condition A. A 31-day Completion Time would be added for restoring the Category A and B limits. This Completion Time is considered acceptable since sufficient battery capacity exists to perform the required function and new Required ACTIONS have been implemented to assure that the battery remains capable of performing its required function (Refer to 02-09-M). This change is consistent with NUREG-1431.
02-09	M	<p>To support the extended Completion Time of new Condition A, within one hour after any Category A or B battery not within limits, the pilot cell electrolyte level and float voltage must be verified within Category B allowable value. This provides a quick indication of the status of the remainder of the battery cells.</p> <p>For Category A and B parameters not restored to within limits, a new requirement would be added to verify the battery meets Category C allowable values once every 7 days.</p> <p>This is consistent with NUREG-1431.</p>
02-10	A	The battery SRs table notes would be revised to add an exception to the requirement to correct specific gravity readings for electrolyte level when the float charge is less than 2 amps. This exception is considered acceptable since charging current provides, in general, a good indication of overall battery condition. This change is consistent with NUREG-1431.
02-11	M	This change to the battery SRs table notes would place a 7-day limit on the use of charging current to satisfy the specific gravity requirements. This is more realistic because CTS does not place a time limit on use of charging current. This change is consistent with NUREG-1431.
02-12	LG	The battery SRs table Note regarding correction of float voltage for average electrolyte temperature would be moved to the Bases. This is consistent with NUREG-1431.
02-13	A	The DC Sources - Shutdown LCO would be revised to require that the DC power subsystem required to be OPERABLE must be the subsystem necessary to support the onsite power distribution-shutdown LCO. This requirement is currently required by CTS [3.8.2.2]. This change is consistent with NUREG-1431 (see also CN 03-06-LS26).
02-14	A	The battery cell parameters table would be revised to add Category C, which now contains the allowable value for each connected cell previously included in Category B. This change is consistent with NUREG-1431.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
02-15	LS4	A Note would be added stating the SRs are not required to be performed for the DC source OPERABILITY in the MODEs governed by DC Sources - Shutdown Applicability. The Note does not delete the requirement that the battery be capable of performing these functions, just that the capacity need not be demonstrated while that battery is relied on to meet the LCO. This change is consistent with NUREG-1431.
02-16	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-17	LS2	The requirements to verify no visible terminal corrosion and minimum electrolyte temperature after a battery discharge or overcharge would be deleted. This is consistent with NUREG-1431.
02-18	LG	The definition of degradation provided in the battery service test SR would be moved to the SR Bases. The number of connected cells that constitutes the ITS SR 3.8.6.3, "representative cells," requirement would be moved to the Bases. This constitutes moving details that are not required to be in TS and therefore is acceptable. This change is consistent with NUREG-1431.
02-19	LS20	Since the DC sources OPERABILITY requirements are in terms of powering DC electrical subsystems required to be OPERABLE by ITS LCO 3.8.10, the Required ACTION would be revised to add, "declare affected required features inoperable," as an alternative to the other Required ACTIONS. In this Condition it may not be necessary to suspend all CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel. In this case, conservative ACTION can be assured if all required equipment without DC power is declared inoperable and the associated ACTIONS taken. Therefore, this proposed change is acceptable from a safety standpoint. This change is consistent with NUREG-1431.
02-20	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-21	LS14	The frequency for battery surveillance would be extended to 24 months if the battery has reached 85 percent of expected life and the rating is \geq 100 percent of manufacturer's capability.
02-22	A	The battery average electrolyte temperature SR would be changed from "above" the minimum temperature to " \geq " the minimum temperature. This change is consistent with NUREG-1431.
02-23	A	The quarterly battery connection resistance SR limit, if there is visible corrosion, would be changed from "less than 150×10^{-6} ohm" to " $\leq 150 \times 10^{-6}$ ohm." This would be consistent with the limit for the 18-month battery corrosion resistance SR. This change is consistent with NUREG-1431.
02-24	LG	The footnote allowing resistance of cell to cell connecting cables to not be included in the connection resistance SR would be moved to the SR Bases. This change is consistent with NUREG-1431.
02-25	LS23	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

DESCRIPTION OF CHANGES TO TS SECTION 3/4.8

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
03-01	LG	The list of required electrical buses, batteries and chargers in the CTS Onsite Power Distribution - Operating [] LCO would be moved to the Bases. This deletes descriptive information from the TS, consistent with NUREG-1431.
03-02	A	Information and requirements for Class 1E inverters would be moved to ITS TS 3.8.7, "Inverters - Operating," ITS 3.8.8, "Inverters - Shutdown," and associated Bases. This change is consistent with NUREG-1431.
03-03	LS20	The Onsite Power Distribution - Shutdown ACTION statement would be revised to allow the option of declaring associated supported required feature(s) inoperable. Although the proposed change allows more operational flexibility, it would still ensure plant safety by taking the appropriate Required ACTION for equipment declared inoperable. Therefore, the change is acceptable. This change is consistent with NUREG-1431.
03-04	M	A second Completion Time for ACTIONS of CTS Onsite Power Distribution - Operating would be added. The Completion Time would place a limit of 16 hours on not meeting the LCO. This Completion Time is required by the NUREG-1431 format that potentially allows alternating between Conditions of ITS LCO 3.8.9 such that the LCO may not be met for an indefinite period of time. This change is consistent with NUREG-1431.
03-05	A	A new ACTION would be added to CTS Onsite Power Distribution - Operating that requires entry into ITS LCO 3.0.3 for two [Class 1E Vital Busses] with inoperable distribution subsystems that result in a loss of safety function. This change is consistent with NUREG-1431.
03-06	LS26	The existing requirement for "the following" distribution buses, lists [] Class 1E power required to be OPERABLE during shutdown . The proposed LCO specifies that the portion of these distribution subsystems necessary to supply AC and DC power to equipment required to be OPERABLE in this plant Condition must be OPERABLE.
03-07	M	An additional ACTION (ITS LCO 3.8.10, Condition A.2.5) would be added to immediately declare associated residual heat removal (RHR) subsystem(s) inoperable and not in operation. This ACTION is included because of the allowances of ITS LCO 3.0.6 (Support/Supported System OPERABILITY). ITS LCO 3.0.6 would allow only the ACTIONS for the inoperable distribution subsystem to be taken. ITS Condition A.2.5 would assure consideration is given to shutdown cooling systems that are without required power and that appropriate ACTIONS are taken to assure OPERABILITY of these required systems. This change is consistent with NUREG-1431.
03-08	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
03-09	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
03-10		Not used.
03-11	A	Not applicable to DCCP. See Conversion Comparison Table Enclosure 3B).
04-01	R	The [Motor-operated Valve Thermal Overload Protection Bypass Devices CTS and] Containment Penetration Conductor Overcurrent Protective Devices CTS would be relocated to licensee-controlled documents . This change is consistent with NUREG-1431.

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(16 pages)

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
01-01 A	<p>TS requirement for minimum DG day tank fuel oil volume and fuel transfer pump would be moved to ITS SR 3.8.1.4 and SR 3.8.1.6.</p> <p>TS requirements for DFO storage would be moved to ITS LCO 3.8.3.</p>	Yes	Yes	Yes	Yes
01-02 A	Footnote would be revised to specify "may" instead of "shall" in reference to the DG start being preceded by an engine prelube period to minimize wear on moving parts that do not get lubricated when the engine is not running.	No, CTS footnote specifies "may."	No, CTS footnote specifies "may."	Yes	Yes
01-03 LS5	TS 3.8.1.1 ACTION requirement to demonstrate DG OPERABILITY with an offsite circuit inoperable, two offsite circuits inoperable, or with an offsite circuit and a DG inoperable, is deleted.	Yes	Yes	No, already deleted per Amendment 101.	No, already deleted per Amendment 112.
01-04 M	This proposed change would revise TS 3.8.1.1 to require that with one or more A.C. electrical power sources inoperable, all A.C. electrical power sources be restored to OPERABLE status within 6 days from the time the first A.C. electrical power source became inoperable.	No, 10-day AOT proposed as Change 01-32-M.	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
01-05 LS6	This proposed change incorporates the NUREG-1431, provision of "determining the OPERABLE DGs are not inoperable due to a common cause failure," which avoids unnecessary testing of OPERABLE DGs, into ACTIONS for one DG inoperable and for one DG and an offsite circuit inoperable. In addition, the requirement to complete DG testing regardless of when the inoperable DG is restored to OPERABLE status would be deleted.	Yes	Yes	Yes	Yes
01-06 LS7	CTS 3.8.1.1 ACTION allowed time for verification of required redundant feature OPERABILITY would be increased from 2 to 4 hours. The requirements would be moved to LCO 3.8.1, ACTION B, and the shutdown requirement would be replaced with a requirement to enter the required feature's ACTION statement.	Yes	Yes	Yes	Yes
01-07 LS3	The CTS SR to start all DGs simultaneously would be revised to delete the requirement to perform this SR during shutdown.	Yes	Yes	No, not in CTS.	No, not in CTS.
01-08 LS8	The CTS 3.8.1.1 ACTION requirement to demonstrate DG operability in 8 hours would be changed to 24 hours.	Yes	Yes	Yes	Yes
01-09	Not Used.	N/A	N/A	N/A	N/A

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
01-10 M	CTS 3.8.1.1 ACTIONS would be revised to require transition to MODE 5 within 30 hours following transition to MODE 3 upon failure to meet the requirements of the ACTION statements.	Yes	No, already in CTS.	Yes	Yes
01-11 TR1	The phrase "or actual" in reference to an automatic actuation signal would be added to those SRs which verify actuation on a test (or simulated) automatic actuation signal.	Yes	Yes	Yes	Yes
01-12 A	The footnote allowing engine prelube period would be applicable to all DG SRs.	Yes	No, already in CTS.	Yes	No, already in CTS.
01-13 LS1	Accelerated testing of the DGs and DG special reporting requirements would be deleted consistent with the recommendations of GL 94-01.	Yes	No, not in CTS.	No, deleted per Amendment 101.	No, deleted per Amendment 112.
01-14 A	CTS requirement of "standby status" would be replaced with "ready-to-load operation."	Yes	Yes	Yes	Yes
01-15 A	The requirement to start the DG from "ambient condition" would be replaced with "standby conditions" and description of these Conditions added to the Bases.	Yes	No, already in CTS.	No, already in CTS.	No, already in CTS.
01-16 LG	The method by which the DG is started would be moved to the Bases.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
01-17 LS10	The requirement to rapidly load the DG in less than or equal to 60 seconds would be deleted.	Yes	No, not in CTS.	No, not in CTS.	No, not in CTS.
01-18 LS11	DG loading requirements would be revised.	Yes	Yes	No, load range already in CTS.	No, not applicable to CTS.
01-19 LS12	A footnote would be added to the several DG loading SRs stating that momentary transients outside the load and/or power factor range do not invalidate the test.	Yes	No, already in CTS.	Yes	Yes
01-20 LG	SRs that support normal design, maintenance, or line-up activities/descriptions would be moved to licensee-controlled documents.	Yes, moved to the Bases.	Yes, moved to TRM.	Yes, moved to USAR.	Yes, moved to the Bases.
01-21 M	An additional requirement would be added to TS 3.8.1.1 ACTION. This requirement would only apply if the inoperable offsite circuit results in no offsite power to a Class 1E electric power distribution system and affected ESF systems.	No, this requirement is not applicable due to plant electrical design.	Yes	Yes	Yes
01-22 M	An additional requirement would be added to TS ACTION for two offsite circuits inoperable. This requirement is intended to provide assurance that an event with a coincident single failure would not result in a complete loss of redundant, required safety functions.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
01-23 LG	The specific loading listed in the DG partial load rejection SR requirement would be replaced by "a load greater than or equal to its single largest post-accident load."	Yes	Yes, moved to Bases.	No, test deleted by Amendment 101.	No, not in CTS per Amendment 112.
01-24 LS13	The DG partial load rejection requirement that voltage be maintained would be replaced with a requirement that within 2.4 seconds following load rejection, the voltage recovers to steady-state requirements.	Yes	No, not in CTS.	No test deleted by Amendment 101.	No, not in CTS.
01-25 M	The DG partial load rejection requirement to maintain frequency at 60 ± 3 Hz would be revised to require that frequency recover to 60 ± 1.2 Hz within 2.4 seconds following load rejection.	Yes	No, not in CTS.	No, test deleted by Amendment 101.	No, not in CTS.
01-26 M	Power factor requirements would be added to the DG loading SRs.	Yes	No, maintain CTS.	No, requirement in CTS, see CN 01-36-A.	No, already in CTS.
01-27 LS9	The SR maximum allowed voltage following DG full load rejection would be increased from 4580 V to 6200 V.	Yes	No, maintains different voltage.	No, maintaining CTS.	No, not applicable to CTS.
01-28 LG	The requirement to verify the DG accelerates within required time would be moved to licensee-controlled documents.	No; CTS requirement retained.	Yes, moved to Bases.	No, not in CTS.	No, already moved per Amendment 112.

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
01-29 A	The SR to verify automatic DG trips are bypassed would be separated from the requirements under simulated SI signal DG testing and included as a separate SR.	Yes	No, different plant design.	No, already a separate CTS SR.	No, already in CTS.
01-30 LG	DG 24-hour load SR voltage and frequency start requirements would be moved to the Bases.	Yes	Yes	Yes	Yes
01-31 A	Cycle-specific requirements that are no longer applicable would be deleted.	Yes	Yes	No, not in CTS.	No, not in CTS.
01-32 M	This proposed change would revise CTS 3.8.1.1 to require that with one or more A.C. electrical power sources inoperable, all A.C. electrical power sources be restored to OPERABLE status within 10 days from the time the first A.C. electrical power source became inoperable.	Yes	No, adopting ITS requirement for 6 days see CN 01-04-M.	No, adopting ITS requirement for 6 days see CN 01-04-M.	No, adopting ITS requirement for 6 days see CN 01-04-M.
01-33 LS15	SR to check for and remove accumulated water from the DG day tanks after each operation of the DG for greater than one hour would be deleted.	Yes	Yes	No, deleted by Amendment 101.	No, deleted by Amendment 112.
01-34	Not used.	N/A	N/A	N/A	N/A
01-35 M	Voltage and frequency requirements would be added to the 10-year simultaneous DG start SR.	Yes	Yes	No, already in CTS.	No, already in CTS.
01-36 A	The DG power factor lower limit of 0.8 would be deleted from the power factor limits of between 0.8 and 0.9.	No, CTS does not contain power factor limits.	No, CTS does not contain power factor limits.	No, maintaining CTS.	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
01-37 LG	The specific requirements for sampling new DFO prior to addition to the storage tanks and for sampling the stored DFO once every 31 days would be moved to the Diesel Fuel Oil Testing Program, which would be referenced by ITS SR 3.8.3.3 and described in ITS 5.5.13.	Yes	Yes	Yes	Yes
01-38 LG	Fuel oil storage tank cleaning requirements would be moved to licensee-controlled documents.	Yes, moved to the FSAR.	Yes, moved to TRM.	No, maintaining CTS.	Yes, moved to the FSAR.
01-39 LG	Fuel oil transfer piping leak test would be moved to licensee controlled documents.	Yes, moved to the FSAR.	Yes, moved to TRM.	No, not in CTS.	No, not in CTS.
01-40 LS25	The requirement for staggered DG testing would be deleted and replaced with a requirement to test each DG every 31 days.	Yes	Yes	Yes	Yes
01-41 M	The AC Sources - Shutdown LCO would be changed to specify that the required AC sources must be powering, or capable of powering, the required AC distribution buses.	Yes	Yes	Yes	Yes
01-42 M	CTS 3.8.1.2, CTS 3.8.2.2, and CTS 3.8.3.2 Applicability would be revised to add "during movement of irradiated fuel assemblies."	Yes	No, maintain CTS.	No, maintain CTS.	No, maintain CTS.
01-43 M	An AC Sources - Shutdown ACTION statement would be revised to specify that the offsite circuit must be capable of supplying power to the required onsite buses, and therefore to all equipment required to be OPERABLE in the Applicable MODEs.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
01-44 LG	Reference to crane operations with loads over the spent fuel pool would be moved from the LCO ACTION to licensee-controlled documents.	Yes, move to the FSAR.	Yes, moved to Bases.	Yes, moved to the USAR.	Yes, moved to the FSAR.
01-45 M	The requirement to initiate corrective ACTION is expanded to match the Applicability of the TS.	Yes	Yes	Yes	Yes
01-46 A	A note would be added to the AC Sources - Shutdown ACTION statement to enter the Distribution Systems LCO if one required train has no electrical power.	Yes	Yes	Yes	Yes
01-47 LS4	The SRs required for AC sources OPERABILITY in MODEs 5 and 6 would be revised to include only those SRs which are applicable. The note listing exceptions to SR required for MODEs 5 and 6 in CTS 4.8.1.2 would be revised to include additional SRs.	Yes	Yes	Yes	Yes
01-48 M	A new LCO would be added with new TS requirements for DG lube oil and starting air.	Yes	Yes	Yes	Yes
01-49 LS16	A new LCO would be added. The Condition of degraded DFO capacity would be addressed by an ACTION statement allowing 48 hours to restore DFO level. A new Condition would address stored DFO with total particulates out of limit and allow 7 days for restoration. A new Condition would address new fuel oil with properties not within limits and allow 30 days for restoration.	Yes	Yes, for DFO capacity only, other changes already in CTS.	Yes	Yes
01-50 LS17	Required features supported by the inoperable DG would be declared inoperable when its required redundant feature is inoperable.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
01-51 LS18	The frequency for testing the DFO transfer pumps would be extended from 31 to 92 days.	No, CTS of 31 days is retained.	Yes	Yes	No, CTS frequency is retained.
01-52 M	SR would require the DFO transfer pumps to be started automatically.	No, DFO pumps started manually.	Yes	No, not a CTS requirement.	No, not a CTS requirement.
01-53 M	A Note would be added to the SR for monthly DG start and load test to allow testing on only one DG at a time.	Yes	Yes	Yes	Yes
01-54 LS19	DG partial load rejection SR frequency lower limit of 53.25 Hz would be deleted.	No, frequency limit not CTS.	Yes	No, test deleted by Amendment 101.	No, not in CTS.
01-55 M	A note would be added prohibiting several SRs from being performed in MODEs 1 - 4.	Yes	Yes	Yes	Yes
01-56 M	The required warmup period prior to hot restart DG SR would be changed from 1 hour to 2 hours.	No, CTS is 2 hour warmup period.	Yes	No, already in CTS.	No, already revised per Amendment 112.
01-57 A	The reference to depressurizing and venting the RCS is removed from this TS. The requirements for providing RCS capacity at low temperatures is covered in the RCS TS Low Temperature Overpressure Protection. Removal of this reference eliminates duplication and is consistent with NUREG-1431.	No, not in CTS.	Yes	No, not in CTS.	No, not in CTS.
01-58 LG	The requirement to verify that the DFO transfer pump transfers fuel from each storage tank to the day tank of each DG via the installed cross-connected lines would be moved to licensee-controlled documents.	No, not CTS requirement.	No, no cross-connected lines.	Yes, moved to the USAR.	Yes, moved to the FSAR.

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
01-59 LS20	AC Sources - Shutdown ACTION statement would be revised to provide the alternative to "declare affected required features with no offsite power available inoperable."	Yes	Yes	Yes	Yes
01-60 LS24	The surveillance interval for verifying that other properties are within limits for ASTM 2D fuel oil is changed from "within 30 days" to "within 31 days" after obtaining a sample.	Yes	Yes	No, not in CTS.	No, not in CTS.
01-61 M	The voltage tolerance would be revised to raise the minimum steady state output voltage requirement for the DGs.	No, CTS already requires adequate minimum voltage.	Yes	No, maintain CTS.	Yes
01-62 A	The verbiage from other applicable ACTIONS for one offsite circuit inoperable and for one DG inoperable would be deleted.	No, ITS format change.	No, alternate markup achieves same result.	Yes	Yes
01-63 LG	The requirement that the air roll test not be performed when the DGs are started per the ACTION requirements has been moved to the Bases.	No, air roll test not in CTS.	Yes	No, not in CTS.	No, not in CTS.
01-64 M	A new LCO requirement would be added with TS requirements for DG turbocharger air assist.	Yes	No, not in CTS.	No, not in CTS.	No, not in CTS.
1-65 A	This change deletes the note which states that "credit may be taken for unplanned events that satisfy this SR." This change eliminates the confusion that may arise with respect to the application of an unplanned event which satisfies the requirements of a given SR by including a discussion in the SR 3.0.1 and Bases. This change is based on Traveler TSTF-8, Rev. 2.	No, Note not in CTS.	No, note not in CTS.	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
1-66 TR3	It is being proposed that this SR be revised to eliminate the requirement to perform the test after any modifications which could affect EDG interdependence.	Yes	Yes	Yes	Yes
02-01 LG	The list of batteries and chargers would be moved to the Bases.	Yes	Yes	Yes	Yes
02-02 A	The phrase, "that could degrade battery performance," would be added to clarify the purpose of the battery inspection SR (TSTF-38).	Yes	Yes	Yes	Yes
02-03 M	The requirement to remove visible terminal corrosion would be added to the SR verifying on an 18-month frequency that cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material.	Yes	Yes	Yes	Yes
02-04 M	A Note would be added to several SRs that this surveillance is not to be performed in MODEs 1, 2, 3, or 4.	Yes	Yes	Yes	Yes
02-05 M	The SR would be changed to allow performing a modified performance discharge test instead of the performance discharge test. The results of the modified performance test could be used in lieu of performing the battery service test, SR 3.8.4.7.	Yes	Yes	Yes	Yes
02-06 LS22	Consistent with industry Traveler TSTF-115, this change would allow the extension of the surveillance frequency verification for battery terminal voltage while on float charge, and for Category A battery cell parameters from 7 days to 31 days in accordance with the recommended frequency of at "least monthly" identified in IEEE 450-1995, Section 4.3.1.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
02-07 A	A note would be added to the battery surveillance requirements table for Category A and B limits. This note would allow the electrolyte level to temporarily increase above the maximum level during equalizing charges provided it does not overflow.	Yes	Yes	Yes	Yes
02-08 LS21	A 31-day Completion Time would be added for restoring the Category A and B limits.	Yes	Yes	Yes	Yes
02-09 M	To support the extended Completion Time of new Condition A, within one hour after any Category A or B battery not within limits, the pilot cell electrolyte level and float voltage must be verified within Category B allowable value. For Category A and B parameters not restored to within limits, a new requirement would be added to verify the battery meets Category C allowable values once every 7 days.	Yes	Yes	Yes	Yes
02-10 A	The battery SRs table notes would be revised to add an exception to the requirement to correct specific gravity readings for electrolyte level when the float charge is less than 2 amps.	Yes	Yes	Yes	Yes
02-11 M	This change to the battery SRs table notes would place a 7-day limit on the use of charging current to satisfy the specific gravity requirements.	Yes	Yes	No, maintaining CTS.	Yes
02-12 LG	The battery SRs table note regarding correction of float voltage for average electrolyte temperature would be moved to the Bases.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
02-13 A	The DC Sources - Shutdown LCO would be revised to require that the DC power subsystem required to be OPERABLE must be the subsystem necessary to support the Onsite Power Distribution - Shutdown LCO.	Yes	Yes	Yes	Yes
02-14 A	The battery cell parameters table would be revised to add Category C, which now contains the allowable value for each connected cell previously included in Category B.	Yes	Yes	Yes	Yes
02-15 LS4	A Note would be added stating SRs that are not required to be performed for the DC source OPERABILITY in the MODEs governed by DC Sources - Shutdown applicability.	Yes	Yes	Yes	Yes
02-16 A	The SR requiring bus breaker alignment and indicated voltage to be verified would be deleted from DC Sources - Shutdown. This requirement would be located in ITS 3.8.9.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
02-17 LS2	The requirements to verify no visible terminal corrosion and minimum electrolyte temperature after a battery discharge or overcharge would be deleted.	Yes	Yes	Yes	Yes
02-18 LG	The definition of degradation provided in the battery service test SR would be moved to the SR Bases. The number of connected cells that constitutes the ITS SR 3.8.6.3, "Representative Cells," requirement would be moved to the Bases.	Yes	Yes	Yes	Yes
02-19 LS20	DC Sources - Shutdown ACTION statement would be revised to allow the option of declaring affected required feature(s) inoperable.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
02-20 A	The battery terminal voltage for Wolf Creek AT&T round cell batteries would be changed per the manufacturer's recommendation.	No	No	Yes	No
02-21 LS-14	The frequency for battery surveillance would be extended to 24 months if the battery has reached 85% of expected life and the rating is \geq 100% of manufacturer's capability.	Yes	Yes	Yes	No, maintaining CTS.
02-22 A	The battery average electrolyte temperature SR would be changed from "above" the minimum temperature to " \geq " the minimum temperature.	Yes	Yes	Yes	Yes
02-23 A	The quarterly battery connection resistance SR limit, if there is visible corrosion, would be changed from "less than 150×10^{-6} ohm" to " $\leq 150 \times 10^{-6}$ ohm."	Yes	Yes	Yes	Yes
02-24 LG	The footnote allowing resistance of cell to cell connecting cables to not be included in the connection resistance SR would be moved to the SR Bases.	Yes	No, not in CTS.	No, not in CTS.	No, not in CTS.
02-25 LS23	Consistent with industry Traveler TSTF-115, this change would allow the performance of a modified discharge test in lieu of a service test at any time.	No, already a part of the CTS.	Yes	Yes	Yes
03-01 LG	The list of required electrical buses, batteries and chargers in the Onsite Power Distribution - Operating [] LCO would be moved to the Bases.	Yes	Yes	Yes	Yes
03-02 A	Information and requirements for Class 1E inverters would be moved to ITS 3.8.7, "Inverters - Operating," and ITS 3.8.8, "Inverters - Shutdown."	Yes	No, not in CTS.	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
03-03 LS20	Onsite Power Distribution - Shutdown ACTION statement would be revised to allow the option of declaring associated supported required feature(s) inoperable.	Yes	Yes	Yes	Yes
03-04 M	A second Completion Time for ACTIONS of Onsite Power Distribution - Operating would be added. The Completion Time would place a limit of 16 hours on not meeting the LCO.	Yes	Yes	Yes	Yes
03-05 A	A new ACTION would be added to Onsite Power Distribution - Operating that requires entry into LCO 3.0.3 for two trains with inoperable distribution subsystems that result in a loss of safety function.	Yes	Yes	Yes	Yes
03-06 LS26	The proposed LCO specifies that the portion of one train of the distribution subsystem necessary to supply AC and DC power to equipment required to be operable in this plant condition must be OPERABLE.	Yes	Yes	Yes	Yes
03-07 M	An additional ACTION (new LCO 3.8.10, Condition A.2.5) would be added to immediately declare associated RHR subsystem(s) inoperable and not in operation.	Yes	Yes	Yes	Yes
03-08 A	An explicit, required 8-hour AOT would be added to the ACTIONS applicable to all portions of the AC electrical power distribution system, not just the 120 volt AC vital system.	No, already part of CTS.	No, already part of CTS.	No, already part of CTS.	Yes
03-09 A	An explicit, required 2-hour AOT would be added to the ACTIONS applicable to the DC electrical power distribution system, not just the 120 volt AC vital system.	No, already part of CTS.	No, already part of CTS.	No, already part of CTS.	Yes

CONVERSION COMPARISON TABLE - CURRENT TS

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
03-10	Not used.	N/A	N/A	N/A	N/A
03-11 A	The Wolf Creek ACTIONS related to inoperable power supplies to emergency service water equipment would be deleted.	No	No	Yes	No
04-01 R	The [Motor-operated Valve Thermal Overload Protection Bypass Devices TS and] Containment Penetration Conductor Overcurrent Protective Devices TS would be relocated to licensee-controlled documents.	Yes, move to Equipment Control Guidelines.	Yes, Containment Penetration Conductor Overcurrent Devices TS relocated to TRM.	No, LCO was previously relocated.	No, LCO was previously relocated.

ENCLOSURE 4
NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)
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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3...." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed changes to delete TS 4.8.1.1.4 and Table 4.8-1, and the associated changes to TS 4.8.1.1.2a. and TS 4.8.1.2 are consistent with the recommendations of NRC GL 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," dated May 31, 1994. With the proposed changes, accelerated testing of the DGs and DG special reporting requirements would be deleted.

Implementation of the maintenance rule for the DGs will continue to ensure DG performance. This program includes requirements to perform detailed root cause analyses of individual DG failures, take corrective actions in response to individual DG failures, and implement DG preventive maintenance actions consistent with the maintenance rule. These changes will enhance the DG reliability. Therefore, the proposed changes will not adversely affect the ability of the DGs to perform their intended safety function.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

Deleting the accelerated DG test frequency and special reporting requirements is consistent with GL 94-01. The provisions of the DCPM Maintenance Program that implements the maintenance rule will continue to ensure DG performance. The program will include requirements to perform detailed root cause analyses of individual DG failures, take corrective actions in response to individual DG failures, and implement DG preventive maintenance actions consistent with the maintenance rule. Deleting special DG reporting requirements is administrative in nature.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes are in accordance with NRC GL 94-01 recommendations. The proposed changes do not involve any physical alterations to the plant and will not affect any analyses assumptions regarding functioning of required equipment designed to mitigate the consequences of accidents.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1
(continued)

3. *Does this change involve a significant reduction in a margin of safety?*

Deletion of the accelerated DG test frequency and special reporting requirements is consistent with GL 94-01. DCP's Maintenance Program will continue to provide assurance of reliable DG performance.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS1" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The requirements to verify no visible terminal corrosion and minimum electrolyte temperature after a battery discharge or overcharge would be deleted. After a battery discharge or overcharge, the requirement to verify that the Category B battery cell parameters are within limits would still be required and provides adequate assurance of battery OPERABILITY.

The requirements to verify temperature and no visible corrosion would continue to be verified every 92 days regardless of battery discharge or overcharge. The 92 day frequency is acceptable for detecting corrosion trends based on operating experience and is consistent with the recommendations of IEEE-450 for frequency of determining the temperature of electrolytes in representative cells. Therefore, the verification of Category B cell parameters following a discharge or overcharge provides adequate assurance of battery OPERABILITY. This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The batteries are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The elimination of these requirements does not affect the battery's capability to perform its required function. Performing verification of Category B battery cell parameters is sufficient to ensure battery OPERABILITY following a discharge or overcharge.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alterations to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed change does not affect the ability of the batteries to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2 (continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently, the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS2" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

CTS SR to start all DGs simultaneously at least once per 10 years or after any modification, which could affect DG starting interdependence, would be revised to delete the requirement to perform the SR during shutdown. This test does not require the DGs to be loaded and paralleled with the offsite power source. Therefore, a grid disturbance could not affect the DGs during performance of this SR. Further, the DGs remain operable during this SR. This change is consistent with NUREG-1431.

Evaluation of this proposed TS revision has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. Deleting the requirement to perform the 10-year DG independence test during shutdown has no effect on the capability of the DGs to perform their intended safety function. The DGs are not required to be paralleled to the offsite grid during this test, and would remain OPERABLE. Deleting this requirement would not effect the design or performance of the DGs.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alterations to the plant. No new failure mechanisms will be introduced by the proposed change. The DGs are designed to provide electrical power to equipment important to safety in the event of a loss of offsite power. The proposed change does not affect the ability of the DGs to start and fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3

(continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently, the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS3" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS4 10CFR50.92 EVALUATION FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The SRs required for AC sources and DC sources OPERABILITY in MODEs 5 and 6 would be revised to include only those which are applicable. In addition, notes would be added stating the SRs that are not required to be performed for OPERABILITY in the MODEs governed by the AC Sources - Shutdown LCO and DC Sources - Shutdown LCO, respectively. SRs were not listed as applicable when shutdown because 1) the SR is only required when DGs are required to be OPERABLE, 2) The SR is only required when the SI signal is OPERABLE, or 3) The SR is only required when the sequencers are required to be OPERABLE. This change is consistent with NUREG-1431.

AC Sources - Shutdown

Many of the currently required surveillances involve tests that would require the one required DG to be paralleled to offsite power; this condition presents a significant risk of a single fault resulting in a station blackout. Other tests, such as load rejection tests, put the availability of the OPERABLE DG at risk during the test. To address this concern and to avoid potential conflicting TS, a note is added to not require that these surveillances be performed in MODEs 5 and 6.

DC Sources - Shutdown

A note would be added stating which SRs are not required to be performed for the DC source OPERABILITY in MODEs 5 and 6. Certain of the currently required SRs involve tests that would cause the battery to be rendered inoperable. If the only required OPERABLE battery were inoperable due to testing, the risk of an event occurring that would require battery operation, would present an additional risk. The exception provided by the note does not exempt the battery from the requirement to be capable of performing the particular function, only that the capability need not be demonstrated while that source of power is being relied upon to support meeting the LCO. This change is consistent with NUREG-1431.

Evaluation of this proposed TS revision has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The electrical power sources support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed SR would continue to provide adequate assurance of the OPERABILITY of the required AC source and DC source functions. The proposed changes would delete the requirement to meet SRs that verify functions which are not required in the applicable MODEs of this TS

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4
(continued)

Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not involve any physical modification to the plant. The proposed changes do not affect the ability of the AC sources or DC sources to fulfill their safety-related function, as required in the applicable MODES of this TS. Hence, no new failure mechanisms will be introduced.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

Elimination of the SRs for AC source functions not required in MODE 5 and 6 does not impact the capability of the AC sources to perform their safety function in these modes. The OPERABILITY of the required AC source and DC source functions would continue to be determined in the same manner.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS4" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The TS 3.8.1.1 ACTION requirements to demonstrate DG OPERABILITY are deleted. ACTIONS currently require all OPERABLE DGs that have not been tested within the previous 24 hours to be started within 24 hours to demonstrate their OPERABILITY in the event that one or both offsite A.C. circuits become inoperable. The intent of this additional testing is to provide added assurance that the remaining OPERABLE DG(s) are capable of supplying emergency power when the offsite sources are degraded.

The normal TS surveillance testing schedule (improved SR 3.8.1.2) demonstrates that OPERABLE DG(s) are capable of performing their intended safety function. The inoperability of one or both of the offsite A.C. circuits does not affect the reliability of the OPERABLE DG(s). Although the occurrence of an inoperable offsite A.C. circuit is infrequent, the present requirements result in an unnecessary test of the DGs, whose reliability is demonstrated by the performance of normal TS surveillance testing. Unnecessary DG testing should be avoided since excessive testing of DGs can cause reduced reliability. Therefore, the requirement to demonstrate OPERABILITY of the OPERABLE DGs whenever an A.C. circuit becomes inoperable would be deleted.

This proposed change is consistent with NRC GL 93-05, "Line-item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing during Power Operation." This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The normal TS surveillance testing schedule (improved SR 3.8.1.2) demonstrates that OPERABLE DG(s) are capable of performing their intended safety function. The inoperability of one or both of the offsite A.C. circuits does not affect the reliability of the OPERABLE DG(s). Deleting this requirement would not effect the design or performance of the DGs.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5 (Continued)

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The DGs are designed to provide electrical power to equipment important to safety in the event of a loss of offsite power. The proposed changes do not affect the ability of the DGs to start and to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. This proposed change should result in an overall improvement in DG reliability and availability due to reduced wear and tear on the DGs by eliminating unnecessary starts. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS5" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

CTS 3.8.1.1 ACTIONS require that the remaining OPERABLE DG(s) be started to demonstrate OPERABILITY in the event a DG becomes inoperable due to any cause other than preventive maintenance or testing. The CTS also requires that the test be completed within 24 hours regardless of when the inoperable DG is returned to OPERABLE status.

The intent of this additional testing is to determine if a common cause failure exists and to provide added assurance that the remaining OPERABLE DG(s) is capable of supplying emergency power. However, this requirement results in unnecessary testing when there is no common cause failure.

This proposed change incorporates the NUREG-1431 provision of "determining the OPERABLE DG(s) are not inoperable due to a common cause failure," which avoids unnecessary testing of OPERABLE DG(s), into ACTION requirements.

Evaluation of this proposed TS revision has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The normal TS surveillance testing schedule (improved SR 3.8.1.2) demonstrates that OPERABLE DGs are capable of performing their intended safety function. The proposed change would provide an alternative to starting the OPERABLE DGs if it can be proven by reasoning and analysis that there is not a common cause failure, thus maintaining assurance the DGs would be OPERABLE to perform their intended safety function if needed. In providing this alternative, unnecessary DG starts may be reduced. Deleting this requirement would not effect the design or performance of the DGs.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The DGs are designed to provide electrical power to equipment important to safety in the event of a loss of offsite power. The proposed changes do not affect the ability of the DGs to start and to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. This proposed change should result in an overall improvement in DG reliability and availability due to reduced wear and tear on the DGs by eliminating unnecessary starts. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS6" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

CTS 3.8.1.1 ACTION requirements allowed time for verification of required redundant feature OPERABILITY would be increased from 2 to 4 hours. The shutdown requirement would be replaced with a requirement to enter the required feature's ACTION statement.

Allowing 4 hours upon discovery of an otherwise OPERABLE required feature powered from an inoperable DG before declaring that required feature inoperable would allow a more reasonable time for repairs to be made prior to declaring the feature inoperable. The additional time proposed will reduce the probability of unnecessary plant transients and plant shutdowns, thus improving plant safety and increasing plant availability.

Increasing the required time to evaluate OPERABILITY of other equipment is consistent with the recommendations of NUREG-1024, "Technical Specifications - Enhancing the Safety Impact." NUREG-1024 states:

Allowable outage times that are too short will subject the plant to unnecessary trips, transients and fatigue cycling. Outage times that are too short also may result in less thorough repair and post-repair testing before equipment is returned to service.

This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. This proposed change would not effect the design or performance of the DGs.

The proposed 4-hour AOT, with entry into the inoperable safety equipment's ACTION statement, takes into account the capacity and capability of the remaining A.C. sources (i.e., the remaining operable offsite circuit(s)), a reasonable time for repairs, OPERABILITY of the redundant counterpart to the inoperable required safety feature, and the low probability of a design basis accident with a loss of offsite power.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7 (Continued)

Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed changes do not affect the ability of the DGs to start and to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS7" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The TS 3.8.1.1 ACTION requirement to demonstrate DG OPERABILITY within 8 hours would be increased to 24 hours. Extending the time required to demonstrate the OPERABILITY of the remaining OPERABLE DGs would allow 24 hours to determine the OPERABLE DGs are not inoperable due to a common cause failure or demonstrate OPERABILITY of the remaining OPERABLE DGs by testing. This proposed change avoids unnecessary DG testing. This change would allow the same time period (24 hours) for determining OPERABILITY of the remaining OPERABLE DG(s) as the ACTION for one inoperable DG currently allows. This proposed change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the DGs.

The normal TS surveillance testing schedule (improved SR 3.8.1.2) demonstrates that OPERABLE DG(s) are capable of performing their intended safety function. The increased Completion Time of 24 hours would allow additional time for a more thorough examination and determination of no potential common cause failure, thus potentially reducing unnecessary DG testing. However, the AOT (12 hours) to return either the DG or offsite circuit would not be changed.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed changes do not affect the ability of the DGs to start and to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8
(Continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS8" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A review of applicable documents (RG 1.108, Rev. 1, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants;" Institute of Electrical and Electronics Engineers (IEEE) Standard 387-1984, "IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations;" and NUREG-1431, Rev. 1) determined that the DG full load rejection test was designed to demonstrate that the DG would not trip on overspeed and that the resulting voltage will not exceed a level that will cause component damage during a full load rejection. A calculation determined that the proposed limit of 6200 volts will not result in component damage and will provide an acceptable voltage limit. Therefore, the proposed change will not adversely affect the ability of the DGs to perform their intended safety function.

Evaluation of this proposed TS revision has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. Thus, increasing the DG full load rejection SR allowed voltage continues to provide adequate assurance that the DGs are OPERABLE following surveillance testing.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alterations to the plant. No new failure mechanisms will be introduced by the proposed change. The DGs are designed to provide electrical power to equipment important to safety in the event of a loss of offsite power. The proposed change does not affect the ability of the DGs to start and fulfill their safety related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9
(continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently, the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS9" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The requirement to rapidly load the DG in less than or equal to 60 seconds would be deleted. Rapid loading of the DG is currently required for the 6-month DG start. Rapidly loading the DG introduces unnecessary mechanical stresses and engine wear.

The purpose of this test is to demonstrate that the DG is capable of accepting loads by simulating the loading rate of the load sequence timers. However, other SRs, the Loss of Offsite Power (LOOP) and LOOP/SI tests, already adequately verify this requirement. Therefore, the SR for rapid loading is unnecessary.

This proposed change is consistent with NRC GL 93-05, "Line-item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing during Power Operation." This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the DGs. The ability of the DGs to rapidly load is demonstrated by other SRs that are performed on a refueling frequency.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed changes do not affect the ability of the DGs to start and to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10
(continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS10" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS11

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The DG loading specified in various SRs would be revised to be consistent with NUREG-1431 requirements. The DG loading requirements would specify a load range that still meets the objective of the loading tests. The load range is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent tear down inspections in accordance with vendor recommendations. This proposed change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the DGs. Providing a load range prevents routine overloading of the DGs, which may result in more frequent tear down inspections.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed changes. The proposed changes do not affect the ability of the DGs to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS11
(continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS11" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS12

10CFR50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A footnote would be added to several SRs stating that momentary transients outside the load range do not invalidate the test, since DG loading could change during this test due to changing bus conditions. Some load fluctuation is expected and should not invalidate this test. The objective of testing controls in these tests is to assure that equilibrium conditions are obtained and maintained, that inadvertent overloads which may damage the machine are avoided and that the test demonstrates the prescribed elements of the machine's ability to carry accident loads and associated power factors. Monitoring/data collection periods during the tests are chosen to assure these criteria are maintained given the inherent margin and stability of the configuration. Therefore, DG load found out of the load range and returned to the band within the monitoring period would not invalidate an DG load test. This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the DGs. Allowing momentary transients outside the load range has no effect on demonstrating that the DG can provide its accident loads with associated power factor at the proper voltage and frequency. This proposed change provides an allowance which recognizes that the DG is paralleled to the offsite power source to supply the required load.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed changes do not affect the ability of the DGs to start and to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS12
(continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. The change does not affect the ability of the test to demonstrate OPERABILITY of the DGs. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS12" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS13

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The DG partial load rejection requirement that voltage be maintained would be replaced with a requirement that within 2.4 seconds following load rejection, the voltage is within steady state requirements. The voltage specified is consistent with the design range of the equipment powered by the DG.

The 2.4 second allowance for voltage and frequency is based on the recommendations of RG 1.9, "Selection and Diesel Generator Set Capacity for Standby Power Supplies," Rev. 3, for recovery from transients caused by the disconnection of the largest single load. This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the DGs. The ability of the DGs to properly recover from rejection of the single largest load will continue to be demonstrated.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed changes do not affect the ability of the DGs to start and to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS13
(continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS13" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS14

10CFR50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS requires a battery discharge performance test every 18 months when the battery has reached 85 percent of its service life. Consistent with NUREG-1431, the proposed change extends the frequency for the battery performance discharge test surveillance to once per 24 months if the battery has reached 85 percent of expected service life and the capacity is ≥ 100 percent of manufacturer's rating.

The proposed TS has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The proposed change extends the frequency for the battery performance discharge test surveillance to once per 24 months if battery has reached 85 percent of expected service life and the capacity is ≥ 100 percent of manufacturer's rating. The surveillance limit of 80 percent capacity is based on meeting the requirements of the applicable safety analyses. This change is acceptable because the battery is still capable of providing its full 100 percent capacity and would not be expected to degrade below the surveillance limit of 80 percent between surveillances. In addition, if the battery were to experience a 10 percent degradation, the 18-month frequency would be required for all following surveillances. Because the capacity of the battery would not be expected to degrade below the acceptance limit between surveillances, this change does not involve a significant increase in probability or consequences of an accident previously evaluated.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed change extends the frequency for the battery surveillance and does not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed changes. The proposed changes do not affect the ability of the battery to fulfill its safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14 (continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS14" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS15

10CFR50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The SR to check for and remove accumulated water from the DG day tanks after each operation of the DG for greater than 1 hour would be deleted. Checking for accumulated water more often than once per 31 days is unnecessarily frequent.

Once per 31 days corresponds to the frequency of the monthly DG test and is adequate to identify and remove any water from the fuel oil system. The main fuel oil storage tanks are checked for water every 31 days and this check would provide additional assurance that a minimal amount of water is present. Cleaning the main fuel oil storage tanks and visual inspection of the accessible DFO transfer piping during operating pressure leak test are required every 10 years and provide additional assurance that water is not introduced to the DFO storage and transfer system. This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the DGs. Deleting the requirement to check for and remove accumulated water from the DG day tank after the DG has operated for greater than or equal to 1 hour has no effect on DG OPERABILITY since presence of water in the DG day tank does not necessarily result in DG failure. This SR will continue to be performed once per 31 days, consistent with normal monthly DG start test and DG day tank fuel oil level test frequency.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed changes do not affect the ability of the DGs to start and to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS15 (continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS15" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS16

10CFR50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

NUREG-1431 LCO 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," includes Conditions and Required ACTIONS that allow reduced DFO inventory for up to 48 hours before requiring that the associated DG be declared inoperable. A new Condition would address stored DFO with total particulates out of limit and allow 7 days for restoration. A new Condition would address new fuel oil with properties not within limits and allow 30 days for restoration.

The Conditions provided for these support systems address the situation where the support system is degraded but still capable of supporting the associated DG. The time allowed by these new TS Conditions to correct the support system problem before declaring the DG inoperable is acceptable based on the remaining capacity or capability of the systems and the low probability of an event occurring during the allowed time that would require the support system's full capacity/capability.

This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the DGs. The changes address the Condition of a support system being degraded but still capable of supporting the associated DG. Therefore, the DG would remain OPERABLE and able to perform its safety function with a degraded support system.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed changes do not affect the ability of the DGs to start and to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16
(continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS16" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS17

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

TS 3.8.1.1 ACTION shutdown requirement would be replaced with a requirement to enter the required feature's ACTION statement. The ACTION currently requires that with one DG inoperable all required safety equipment that depends on the remaining OPERABLE DG(s) be verified OPERABLE. If these requirements are not met, a unit shutdown is required. Rather than requiring a unit shutdown, the proposed change would require declaring inoperable the required safety equipment powered from an inoperable DG. This would result in entering the required feature's TS ACTION statement.

This Required ACTION provides assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of a critical system. The features covered by this action are designed with redundant safety-related trains. Single train systems, such as the turbine-driven auxiliary feedwater pump, are not included.

For systems where a loss of redundant features is not as significant (i.e., some time allowed prior to the requirement to shut down), recognition of the difference is appropriate and would be recognized.

This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The DGs are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the DGs. By declaring the required feature inoperable when its redundant required feature is inoperable and entering the ACTION requirement, the assumed OPERABILITY of safety-related equipment is maintained. This Required ACTION provides assurance that a postulated loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of a critical system.

V. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS17 (continued)

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed changes do not affect the ability of the DGs to start and to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS17" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS20

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Since the AC sources and DC sources OPERABILITY requirements, ITS LCOs 3.8.2, 3.8.5, and 3.8.8 are in terms of powering AC and DC electrical subsystems required to be OPERABLE by LCO 3.8.10, the Required ACTION for these TS would be revised to declare affected required features inoperable, as an alternative to the other Required ACTIONS. In this Condition it may not be necessary to suspend all CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel. In this case, conservative ACTION can be assured if all required equipment without the required power source (i.e., AC or DC power) is declared inoperable and the associated ACTIONS taken.

In the Condition of LCO 3.8.2, with one or more required load centers, motor control centers, buses, etc., not capable of being powered by an offsite circuit, it may not be necessary to perform ITS 3.8.2 Required ACTIONS A.2.1 through A.2.4. Plant safety would be assured by declaring inoperable all required equipment without qualified offsite power and taking the specified ACTIONS. The requirements for LCO 3.8.5 and LCO 3.8.8 would be similar.

This proposed change would provide additional flexibility in responding to an inoperable offsite circuit or inoperable DC source while in MODE 5 or 6. This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The electrical power sources are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change would not effect the design or performance of the electrical power sources. The allowance of declaring the required feature inoperable ensures a commensurate level of plant safety as the ACTIONS required with one offsite circuit inoperable or with one or more DC electrical power subsystem inoperable.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS20 (continued)

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed changes do not affect the ability of the electrical power systems to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS20" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS21

10CFR50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Notes contained in current battery SRs table for Category A and B limits would be combined in new LCO 3.8.6, "Battery Cell Parameters," as Condition A. A 31-day Completion Time would be added for restoring the Category A and B limits. This Completion Time is considered acceptable since sufficient battery capacity exists to perform the required function.

To support this extended Completion Time, two new requirements have been added. ITS Required ACTION A.1. requires that pilot cell electrolyte level and float voltage are within Category C limits within 1 hour whenever the Category A or B limits are not met. This proposed Required ACTION provides a quick indication of the status of the remainder of the battery cells. Required ACTION A.2., performed within 24 hours of entry into Condition A and once per 7 days thereafter, verifies battery cell parameters for all the cells are within the Category C limits. The Category C limits are the limits beyond which the battery is considered immediately inoperable. Performing Required ACTION A.2. provides assurance that the battery is still capable of performing its required function. If Category C limits are not met or the Category A and B limits are not restored within 31 days, ITS ACTION B. requires the affected battery to be declared inoperable.

This change is consistent with NUREG-1431.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The batteries are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any previously analyzed accident. The proposed change allows 31 days for restoration of battery cell parameters, provided Category C parameters are met. The proposed change will not allow continuous operation when sufficient battery capacity does not exist (i.e., Category C limits not met). The increased time allowed to continue operation with degraded batteries prior to requiring the batteries be declared inoperable is acceptable based on the low probability of a DBA requiring the DC power sources to perform their required function, considering the desire to minimize unnecessary plant transients.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS21 (continued)

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed changes do not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed changes do not affect the ability of the batteries to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed changes will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS21" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS22

10CFR50.92 EVALUATION
FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with industry Traveler TSTF-115, this change would allow the extension of the surveillance frequency for battery terminal voltage while on float charge, and for Category A battery cell parameters from 7 days to 31 days in accordance with the recommended frequency of at "least monthly" identified in IEEE-450-1995, Section 4.3.1. CTS [SRs 4.8.3.1a.1 and 4.8.2.1] require a once per 7 days frequency for surveillance verification and this proposed change would extend the surveillance frequency to 31 days consistent with [manufacturer's recommendations and] IEEE-450-1995.

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the Condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state. Category A defines the normal parameter limit for each designated pilot cell in each battery. The cell selected as a pilot cell are those whose temperature, voltage, and electrolyte specific gravity approximate the Condition of the entire battery. The Category A limits specified are based on [manufacturer's recommendations and] IEEE-450-1995.

This proposed change would provide additional flexibility in allowing the extension of the surveillance frequency verifications for battery terminal voltage and Category A battery cell parameters from 7 to 31 days and the change is consistent with IEEE-450-1995.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The batteries are used to support mitigation of the consequences of an accident, and are not considered to be an initiator of any accident. The proposed change would not effect the design or performance of the batteries. The allowance to extend the surveillance frequency from 7 days to 31 days is consistent with the recommended frequency of at least monthly identified in IEEE-450-1995.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS22 (continued)

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed change does not involve any physical alterations to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed change does not affect the ability of the batteries to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed change will not alter any accident analysis assumptions, initial Conditions, or results. Consequently the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS22" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS24

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, the surveillance interval for verifying that other DFO properties are within limits for ASTM 2D fuel oil is changed from "within 30 days" to "within 31 days" after obtaining a sample. The fuel properties that can have an immediate detrimental impact on diesel combustion, (i.e., API gravity, kinematic viscosity, flash point and appearance) are verified prior to addition to the storage tank. The "other properties" may be analyzed after addition to the tank. The 31-day verification interval for these properties is acceptable because the fuel properties of interest, even if they are not within their stated limits, would not have an immediate effect on DG operation. The CTS 30-day verification interval was probably chosen because it was a convenient time interval for sending the sample and receiving the results from the laboratory selected for testing. NUREG-1431 has selected a 31-day testing interval. The 1 day increase in the interval would not have a significant effect on the acceptability of the DFO.

This proposed change would provide additional flexibility in allowing the performance of a modified performance discharge test in lieu of a service test at any time and the change is consistent with IEEE-450-1995.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The proposed change affects the interval for the verification of certain properties of diesel fuel. This testing interval does not impact an accident initiator and thus cannot increase the probability of an accident. The DGs are used to mitigate the consequences of accidents. However, even if the fuel properties of interest are not within their stated limits, the fuel would not have an immediate effect on DG operation. Thus, the diesel could continue to perform its safety function.

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed change does not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change. The proposed change does not affect the ability of the diesels to fulfill their safety-related function.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS24
(continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The proposed change will not alter any accident analysis assumptions, initial Conditions, or results. Consequently, the proposed change does not have any effect on the margin of safety.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "LS24" resulting from the conversion to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS25 10 CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would eliminate the requirement to test multi-train/multi-component systems on a STAGGERED TEST BASIS (STB) while maintaining the testing frequency for individual trains/components unchanged. As discussed in the NRC's safety evaluation report for Plant Vogtle's TS conversion, "the intent of a requirement for staggered testing is to increase the reliability of the component or system being tested. A number of studies have demonstrated that staggered testing has negligible impact on component reliability. These analytical and deterministic studies have shown that in most cases staggered testing (a) is operationally difficult, (b) has negligible impact on component reliability, (c) is not as safety significant as initially thought, (d) introduces additional stress on components such as DG potentially causing increased component failure rates and component wear out, (e) results in more frequent reductions in system redundancy for testing purposes, and (f) increases the likelihood of human error by increasing the number of separate test evolutions. Therefore, changes of this type are acceptable. Accordingly, many of the staggered testing requirements in the CTS have been omitted from the improved TS." Based on the NRC safety evaluation, removal of STB testing would have negligible negative impact, and potential positive impact, on plant safety.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change would not have an appreciable effect any accident initiators or precursors. The change involves the testing frequency of accident mitigating systems, and therefore would not significantly effect the probability of occurrence of previously evaluated accidents. Also, the change would have a negligible impact on the availability of accident mitigating systems, so the consequences of any previously evaluated accidents would not be significantly increased.

Therefore, the proposed change would have no significant effect on the probability or consequences of any previously analyzed accidents.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes would not involve any physical alteration to the plant or systems or change the manner in which any safety-related system performs its safety function.

Therefore, the proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS25
(continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change would not alter any accident analysis assumptions, initial Conditions, or results. Consequently it would not have any effect on margins of safety.

Therefore, the proposed change would have no significant adverse effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS25" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS26
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change revises the requirement for OPERABLE onsite shutdown power. The CTS requires that 1 [bus] (subsystem) of the various power supplies and buses be OPERABLE. The change requires that only the necessary portion of these subsystems be OPERABLE. The necessary portion are those portions required to support the equipment on that [bus] which is required to be OPERABLE in the existing shutdown Condition. This change is consistent with NUREG-1431 and is acceptable because there is no reason to have portions of the power systems OPERABLE that are not supporting components which are being credited in the safety analysis for shutdown events.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change does not involve any physical changes to the plant. The change does change the way the plant is operated in that portions of the power system which are not supporting required equipment are no longer required to be OPERABLE. This change in the way the plant is operated has no impact on the probability of any of the accidents assumed during plant shutdown and will not impact the consequences of these accidents since the credited equipment will continue to be supported by power sources as they were prior to the change.

Therefore, this change does not involve a significant increase the probability or consequences of an accidents previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS26
(continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not involve any physical alteration to the plant. No new failure mechanisms will be introduced by the proposed change.

Therefore, the proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not alter any accident analysis assumptions, initial Conditions, or results. The equipment which is credited in the accident analyses will be powered in the same manner as before the change. Consequently, the proposed change does not have any effect on the margin of safety.

Therefore, the proposed change would have no significant adverse effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, it is concluded that the activities associated with NSHC "LS26" resulting from the conversion to the improved TS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

V. RECURRING NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC TR1

10CFR50.59 EVALUATION

FOR

RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the CTS to allow the use of actual actuation signals for SRs that currently require testing using simulated test signals only. This change is consistent with NUREG-1431.

In several SRs in the CTS, OPERABILITY of certain equipment is demonstrated by ensuring that the equipment performs its safety function upon receipt of a simulated test signal. The intent of a 'simulated' signal was to be able to perform the required testing without the occurrence (or without causing) an actual signal generating event. However, the unintended effect was to require the performance of the SR (using a test signal) even if an actual signal had previously verified the operation of the equipment. This change allows credit to be taken for actual events when the required equipment actuates successfully.

While the occurrence of events that cause actuation of accident mitigation equipment is undesirable, the actuation of mitigation equipment on an actual signal is a better demonstration of its OPERABILITY than an actuation using a test signal. Thus, the change does not reduce the reliability of the equipment tested. The change also improves plant safety by reducing the amount of time the equipment is taken out of service for testing, and thereby increasing its availability during an actual event and by reducing the wear of the equipment caused by unnecessary testing.

Evaluation of this proposed TS has determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety.*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The proposed change allows the use of an actual actuation signals (when/if it occurs) to satisfy SRs currently requiring simulated test signals to demonstrate equipment OPERABILITY. While the change takes advantage of events that may have occurred, it has no adverse effect on any accident initiators or accident consequences. In fact, by potentially reducing unnecessary testing, it may reduce the probability of an accident because the testing itself can increase the probability of an accident. It may also reduce accident consequences by increasing the equipment availability (i.e., less time in test).

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR1 (continued)

Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

The use of an actual actuation signal to satisfy an SR has either no impact on, or increases the margin of plant safety by:

- a) Increasing mitigation equipment availability, and
- b) Improving mitigation equipment reliability by potentially reducing wear caused by unnecessary testing.

The change is consistent with the safety analysis and licensing basis.

Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, the activities associated with NSHC "TR1" resulting from the conversion of CTS 3/4.8 to the ITS format are concluded to satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR2

10 CFR 50.92 EVALUATION
FOR

RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change in accordance with NUREG-1431, Rev. 1, removes the requirement for a special report to be generated and submitted to the NRC. Reporting to the NRC will be done commensurate with the reporting requirements of 10CFR 50.72 and 50.73.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

This change is purely an administrative reporting change and cannot affect any accident probability or consequences.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

This change is purely an administrative reporting change and cannot create any new accident or affect any accident previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

This change is purely an administrative reporting change and does not affect any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "TR2" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

V. RECURRING NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR3

10 CFR 50.92 EVALUATION FOR RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision is to remove reference to specific post-maintenance tests from the ITS. Post-Maintenance Testing Programs are controlled via plant administrative procedures in accordance with licensee controlled document (ITS Section 5.4.1, Procedures commitments to NRC RG 1.33, "Quality Assurance Program Requirements (Operation)," and ANS 3.2-ANSI N18.7, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." Specific post-maintenance testing requirements are contingent on the type and scope of maintenance actually performed as well as the availability and viability of test equipment, techniques, etc. Removal of specific testing requirements from the ITS and reliance on normal post-maintenance testing programs addressed by licensee controlled documents allow flexibility to modify testing to address the circumstances of the maintenance performed while still assuring OPERABILITY of equipment returned to service,

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of a accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

No, this is an administrative change which removes specific post-maintenance test requirements from TS. The testing, or equivalent testing, to assure equipment OPERABILITY prior to return to service would still be done as required by normal plant maintenance retest programs. Therefore, this change would not result in any increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

No, this is an administrative change and does not create a new or different kind of accident from any previously evaluated.

3. *Does this change involve a significant reduction in a margin of safety?*

No. This change is an administrative change and does not affect any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "TR3" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers (1 Page)

NUREG-1431 Specifications that are not applicable (1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
3.8.1	3.8-1
3.8.2	3.8-20
3.8.3	3.8-24
3.8.4	3.8-30
3.8.5	3.8-34
3.8.6	3.8-37
3.8.7	3.8-42
3.8.8	3.8-44
3.8.9	3.8-47
3.8.10	3.8-49

Methodology (2 Pages)

Industry Travelers Applicable to Section 3.4/8

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-2, Rev. 1	Incorporated	3.8-31	
TSTF-8, Rev 2	Incorporated	3.8-20	NRC approved.
TSTF-16, Rev. 1	Not Incorporated	N/A	Not NRC approved as of traveler cut-off date.
TSTF-36, Rev. 2	incorporated	3.8-35	DCPP only.
TSTF-37, Rev. 1	Incorporated	3.8-05	
TSTF-38	Incorporated	3.8-41	NRC approved.
TSTF-51	Not Incorporated	N/A	Requires plant-specific reanalysis to establish decay time dependence for fuel handling accident.
TSTF-115	Partially incorporate	3.8-34, 3.8-38	May be re-issued under new TSTF number.
TSTF-163	Not incorporated	N/A	Not NRC approved as of traveler cut-off date.

NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

None

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

- a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; ~~and~~ B
- b. ~~Three~~ Two diesel generators (DGs) capable of supplying the onsite Class 1E power distribution subsystem(s); ~~and~~ 3.8-01
- c. ~~Two supply trains of the diesel fuel oil (DFO) transfer system Automatic load sequencers for Train A and Train B.~~ 3.8-04
B-PS

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One required offsite circuit inoperable.</p>	<p>A.1 Perform SR 3.8.1.1 for required OPERABLE offsite circuit.</p> <p><u>AND</u></p> <p>A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.</p> <p><u>AND</u></p>	<p>1 hour <u> </u> <u> </u> B <u> </u></p> <p><u>AND</u></p> <p>Once per 8 hours thereafter</p> <p>24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)</p> <p><u> </u> 3.8-32 <u> </u></p> <p>(continued)</p>

ACTIONS		
CONDITION	REQUIRED ACTION	COMPLETION TIME
(continued)	A-3A.2 Restore required offsite circuit to OPERABLE status.	72 hours <u>3.8-32</u> AND <u>B</u> <u>3.8-02</u> 610 days from discovery of failure to meet LCO
B. One [required] DG inoperable.	<p>B.1 Perform SR 3.8.1.1 for the required offsite circuit(s).</p> <p>AND</p> <p><u>NOTE</u> In MODE 1, 2, and 3, TDAFW pump is considered a required redundant feature.</p> <p>B.2 Declare required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable.</p> <p>AND</p> <p>B.3.1 Determine OPERABLE DG(s) is not inoperable due to common cause failure.</p> <p>OR</p> <p>B.3.2 Perform SR 3.8.1.2 for OPERABLE DG(s).</p> <p>AND</p>	<p>1 hour <u>B-PS</u></p> <p>AND</p> <p>Once per 8 hours thereafter <u>3.8-43</u></p> <p>4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)</p> <p>24 hours <u>B</u></p> <p>24 hours <u>B</u></p> <p>(continued)</p>

ACTIONS		
CONDITION	REQUIRED ACTION	COMPLETION TIME
(continued)	B.4 Restore [required] DG to OPERABLE status.	7 days 72 hours PS <u>B-PS</u> <u>3.8-03</u> AND 10 6 day S from discovery of failure to meet LCO <u>3.8-02</u>
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. AND C.2 Restore one required offsite circuit to OPERABLE status.	B 12 hours from discovery of Condition C concurrent with inoperability of redundant required features <u>B</u> 24 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. One required offsite circuit inoperable.</p> <p><u>AND</u></p> <p>One [required] DG inoperable.</p>	<p><u>NOTE</u></p> <p>Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems Operating," when Condition D is entered with no AC power source to any train.</p> <p>D.1 Restore required offsite circuit to OPERABLE status.</p> <p><u>OR</u></p> <p>D.2 Restore [required] DG to OPERABLE status.</p>	<p>B</p> <p>3.8-32</p> <p>B-PS</p> <p>12 hours</p> <p>B</p> <p>B-PS</p> <p>12 hours</p>
<p>E. Two or more [required] DGs inoperable.</p>	<p>E.1 Restore Ensure at least two one [required] DGs to are OPERABLE status.</p>	<p>2 hours</p> <p>3.8-01</p> <p>B-PS</p> <p>(continued)</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>F. One supply train of the DFO transfer system inoperable.</p> <p>F. REVIEWER'S NOTE This Condition may be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads following a loss of offsite power independent of, or coincident with, a Design Basis Event.</p> <p>One [required] [automatic load sequencer] inoperable.</p>	<p>F.1 Restore the DFO transfer system to OPERABLE status.</p> <p>F.1 Restore [required] [automatic load sequencer] to OPERABLE status.</p>	<p>72 hours 3.8-04 3.8-01</p> <p>[12] hours</p>
<p>G. Two supply trains of the DFO transfer system inoperable.</p>	<p>G.1 Restore one train of the DFO transfer system to OPERABLE status.</p>	<p>1 hour 3.8-04 3.8-01</p>
<p>H G. Required Action and associated Completion Time of Condition A, B, C, D, E, F or G not met.</p>	<p>H G.1 Be in MODE 3.</p> <p>AND</p> <p>H G.2 Be in MODE 5.</p>	<p>6 hours 3.8-04</p> <p>36 hours B</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>IH. Three or more [required] AC sources inoperable.</p> <p>Two or more DGs inoperable.</p> <p>AND</p> <p>One or more required offsite circuits inoperable.</p>	<p>IH.1 Enter LCO 3.0.3.</p>	<p>Immediately</p> <p><u>B-PS</u></p> <p><u>3.8-01</u></p>
<p>0 One or more DGs inoperable.</p> <p>AND</p> <p>Two required offsite circuits inoperable.</p>	<p>0 1 Enter LCO 3.0.3.</p>	<p>Immediately</p> <p><u>3.8-01</u></p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.3 -----NOTES-----</p> <ol style="list-style-type: none"> 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.7. <p>-----</p> <p>Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2370 [4500] kW and ≤ 2610 [5000] kW.</p>	<p style="text-align: right;"><u>3.8-05</u></p> <p style="text-align: right;"><u>B-PS</u></p> <p>31 days As specified in Table 3.8.1-1</p>
<p>SR 3.8.1.4 Verify each day tank [and engine mounted tank] contains ≥ 250 [220] gal of fuel oil.</p>	<p>31 days <u>B-PS</u></p> <p><u>B-PS</u></p>
<p>SR 3.8.1.5 Check for and remove accumulated water from each day tank [and engine mounted tank].</p>	<p>31 days <u>B-PS</u></p> <p>days <u>B</u></p>
<p>SR 3.8.1.6 Verify the fuel oil transfer system operates to [automatically] transfer fuel oil from storage tanks to the day tank [and engine mounted tank].</p>	<p>31 days <u>B-PS</u></p> <p><u>B-PS</u></p> <p><u>B-PS</u></p> <p style="text-align: right;">(continued)</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.7 -----NOTE----- All DG starts may be preceded by an engine prelude period. ----- Verify each DG starts from standby condition and achieves: a. in ≤ 10 seconds, speed ≥ 900 rpm, and b. in ≤ 13 seconds, voltage ≥ 3785 [3740] V and ≤ 4400 [4500] V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.</p>	<p>184 days</p> <p><u>3.8-40</u></p> <p><u>B-PS</u></p> <p><u>B-PS</u></p> <p><u>B</u></p>
<p>SR 3.8.1.8 -----NOTE----- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. ----- Verify automatic and manual transfer of AC power sources from the normal offsite circuit to the each alternate required offsite circuit and manual transfer from the alternate offsite circuit to the delayed access circuit.</p>	<p><u>3.8-20</u></p> <p><u>B</u></p> <p><u>B</u></p> <p><u>3.8-06</u></p> <p>18 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.9 -----NOTES-----</p> <p>1. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR.</p> <p>2. If performed with the DG synchronized with offsite power, it shall be performed at a power factor \leq {0.9}.</p> <p>-----</p> <p>Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and:</p> <p>a. Following load rejection, the frequency is \leq 63 Hz;</p> <p>b. Within 2.4 {3} seconds following load rejection, the voltage is \geq 3785 {3740} V and \leq 4400 {4500} V; and</p> <p>c. Within 2.4 {3} seconds following load rejection, the frequency is \geq 58.8 Hz and \leq 61.2 Hz.</p>	<p>3.8-20</p> <p>B</p> <p>B</p> <p>B</p> <p>18 months</p> <p>B</p> <p>B-PS</p> <p>B-PS</p> <p>B</p>
<p>SR 3.8.1.10 -----NOTE-----</p> <p>This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR.</p> <p>-----</p> <p>Verify each DG operating at a power factor \leq 0.87 {0.9} does not trip and voltage is maintained \leq 6200 {5000} V during and following a load rejection of \geq 2370 {4500} kW and \leq 2610 {5000} kW.</p>	<p>3.8-20</p> <p>B</p> <p>B</p> <p>18 months</p> <p>B-PS</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.11 -----NOTES-----</p> <ol style="list-style-type: none"> 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. <p>-----</p> <p>Verify on an actual or simulated loss of offsite power signal:</p> <ol style="list-style-type: none"> a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: <ol style="list-style-type: none"> 1. energizes permanently connected loads in ≤ 10 seconds. 2. energizes auto-connected shutdown loads through auto-transfer sequencing timers. 3. maintains steady state voltage ≥ 3785 [3740] V and ≤ 4400 [4580] V. 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected shutdown loads for ≥ 5 minutes. 	<p><u>3.8-20</u></p> <p><u>18 months</u></p> <p><u>B</u></p> <p><u>B</u></p> <p><u>B-PS</u></p> <p><u>B-PS</u></p> <p><u>B</u></p> <p><u>B</u></p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.12 -----NOTES-----</p> <p>1. All DG starts may be preceded by an engine prelube period.</p> <p>2. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR.</p> <p>-----</p>	<p>ED</p> <p>3.8-20</p>
<p>Verify on an actual or simulated Safety Injection Engineered Safety Feature (ESF) actuation signal each DG auto-starts from standby condition and:</p> <p>a. In ≤ 13 [10] seconds after auto-start and during tests, achieves voltage ≥ 3785 [3740] V and ≤ 4400 [4580] V;</p> <p>b. In ≤ 13 [10] seconds after auto-start and during tests, achieves frequency ≥ 58.8 Hz and ≤ 61.2 Hz;</p> <p>c. Operates for ≥ 5 minutes;</p> <p>d. Permanently connected loads are remain energized from the alternate offsite power source system; and</p> <p>e. Emergency loads are energized or auto-connected through the automatic ESF load sequencing timers sequencer from to the alternate offsite power source system.</p>	<p>B</p> <p>18 months</p> <p>B-PS</p> <p>B-PS</p> <p>B</p> <p>3.8-06</p> <p>3.8-06</p> <p>B-PS</p> <p>3.8-06</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.13 -----NOTE----- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. ----- Verify each DG's automatic trips are bypassed when the diesel engine trip cutout switch is in the cutout position and the DG is aligned for automatic operation on [actual or simulated loss of voltage signal on the emergency bus concurrent with an actual or simulated ESF actuation signal] except:</p> <ul style="list-style-type: none"> a. Engine overspeed; b. Generator differential current; and c. Low lube oil pressure; d. [High crankcase pressure;] and e. [Start failure relay]. 	<p><u>3.8-20</u> <u>B</u> <u>18 months</u> <u>B</u> <u>B-PS</u> <u>B</u> <u>B</u> <u>B-PS</u></p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.14 -----NOTES-----</p> <p>1. Momentary transients outside the load and power factor ranges do not invalidate this test.</p> <p>2. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR.</p> <p>-----</p> <p>Verify each DG operating at a power factor ≤ 0.87 [0.9] operates for ≥ 24 hours:</p> <p>a. For ≥ 2 hours loaded ≥ 2625 [5250] kW and ≤ 2890 [5500] kW; and</p> <p>b. For the remaining hours of the test loaded ≥ 2370 [4500] kW and ≤ 2610 [5000] kW.</p>	<p>3.8-20</p> <p><u>B</u></p> <p>18 months</p> <p><u>B-PS</u></p> <p><u>B</u></p> <p><u>B-PS</u></p> <p><u>B-PS</u></p>
<p>SR 3.8.1.15 -----NOTES-----</p> <p>1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 2 hours loaded ≥ 2370 [4500] kW and ≤ 2610 [5000] kW.</p> <p>Momentary transients outside of load range do not invalidate this test.</p> <p>2. All DG starts may be preceded by an engine prelube period.</p> <p>-----</p> <p>Verify each DG starts and achieves:</p> <p>a. In ≤ 10 seconds, speed ≥ 900 rpm, and</p> <p>b. in ≤ 13 [10] seconds, voltage ≥ 3785 [3740] V, and ≤ 4400 [4580] V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.</p>	<p><u>B</u></p> <p><u>B-PS</u></p> <p>18 months</p> <p>3.8-40</p> <p><u>B</u></p> <p><u>B-PS</u></p> <p><u>B</u></p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.16 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. ----- Verify each DG:</p> <ul style="list-style-type: none"> a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation. 	<p><u>3.8-20</u></p> <p><u>18 months</u></p> <p><u>B</u></p>
<p>SR 3.8.1.17 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. ----- Verify, with a DG operating in test mode and connected to its bus, an actual or simulated Safety Injection ESF actuation signal overrides the test mode by:</p> <ul style="list-style-type: none"> a. Returning DG to ready to load operation [; and b. Automatically energizing the emergency load from offsite power]. a. Opening the auxiliary transformer breaker; and b. Automatically sequencing the emergency loads onto the DG. 	<p><u>3.8-20</u></p> <p><u>B</u></p> <p><u>18 months</u></p> <p><u>B</u></p> <p><u>B-PS</u></p> <p><u>3.8-08</u></p>

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.19 (continued)</p> <ol style="list-style-type: none"> 2. energizes auto-connected emergency loads through load sequencing timer sequencer. 3. achieves steady state voltage ≥ 3785 [3740] V and ≤ 4400 [4500] V, 4. achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes. 	<p><u>3.8-09</u></p> <p><u>B-PS</u></p> <p><u>B</u></p> <p><u>B</u></p>
<p>SR 3.8.1.20 -----NOTE----- All DG starts may be preceded by an engine prelube period. -----</p> <p>Verify when started simultaneously from standby condition, each DG achieves:</p> <p>a. in ≤ 10 seconds, speed ≥ 900 rpm, and</p> <p>b. in ≤ 13 [10] seconds, voltage ≥ 3785 [3744] V and ≤ 4400 [4576] V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.</p>	<p>10 years</p> <p><u>3.8-40</u></p> <p><u>B-PS</u></p> <p><u>B-PS</u></p> <p><u>B</u></p>

Table 3.8.1-1 (page 1 of 1)

~~Diesel Generator Test Schedule~~

NUMBER OF FAILURES IN LAST 25 VALID TESTS ^(a)	FREQUENCY
≤ 3	31 days
≥ 4	7 days ^(b) (but no less than 24 hours)

~~(a) Criteria for determining number of failures and valid tests shall be in accordance with Regulatory Position C.2.1 of Regulatory Guide 1.9, Revision 3, where the number of tests and failures is determined on a per DG basis.~~

3:8-05

~~(b) This test frequency shall be maintained until seven consecutive failure free starts from standby conditions and load and run tests have been performed. This is consistent with Regulatory Position [], of Regulatory Guide 1.9, Revision 3. If, subsequent to the 7 failure free tests, 1 or more additional failures occur, such that there are again 4 or more failures in the last 25 tests, the testing interval shall again be reduced as noted above and maintained until 7 consecutive failure free tests have been performed.~~

~~Note: If Revision 3 of Regulatory Guide 1.9 is not approved, the above table will be modified to be consistent with the existing version of Regulatory Guide 1.108, GL 84 15, or other approved version.~~

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2

The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown"; and

b. One diesel generator (DG) capable of supplying ~~one train of~~ the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10-
and

3.8-15

~~c. One supply train of the diesel fuel oil (DFO) transfer system~~

3.8-04

APPLICABILITY:

MODES 5 and 6.

During movement of irradiated fuel assemblies.

ACTIONS

NOTE
 LCO 3.0.3 is not applicable

3.8-35

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One required offsite circuit inoperable.</p>	<p>-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train <u>Class 1E AC electrical power distribution subsystem</u> de-energized as a result of Condition A. -----</p> <p>A.1 Declare affected required feature(s) with no offsite power available inoperable.</p> <p><u>OR</u></p> <p>A.2.1 Suspend CORE ALTERATIONS.</p> <p><u>AND</u></p>	<p><u>3.8-15</u></p> <p>Immediately</p> <p>Immediately</p> <p>(Continued)</p>

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. (continued)</p>	<p>A.2.2 Suspend movement of irradiated fuel assemblies. <u>AND</u> A.2.3 Initiate action to suspend operations involving positive reactivity additions. <u>AND</u> A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status.</p>	<p>Immediately Immediately Immediately</p>
<p>B. One The required DG inoperable. <u>OR</u> The required supply train of the DFO transfer system inoperable.</p>	<p>B.1 Suspend CORE ALTERATIONS. <u>AND</u> B.2 Suspend movement of irradiated fuel assemblies. <u>AND</u> B.3 Initiate action to suspend operations involving positive reactivity additions. <u>AND</u> B.4 Initiate action to restore required DG to OPERABLE status.</p>	<p>Immediately Immediately <u>3.8-04</u> Immediately Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.2.1 -----NOTE----- The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.9 through SR 3.8.1.11, SR 3.8.1.13, SR 3.8.1.14 through SR 3.8.1.16, and SR 3.8.1.18 (for auto transfer timers). ----- For AC sources required to be OPERABLE, the following SRs of Specification 3.8.1, "AC Sources - Operating," except SR 3.8.1.8, SR 3.8.1.12, SR 3.8.1.16, SR 3.8.1.17, SR 3.8.1.18 (for ESF timers), SR 3.8.1.19, and SR 3.8.1.20, are applicable: SR 3.8.1.1 through SR 3.8.1.7, SR 3.8.1.9 through SR 3.8.1.11, SR 3.8.1.14 through SR 3.8.1.16, and SR 3.8.1.18 (for auto transfer timers).</p>	<p style="text-align: center;"><u>3.8-17</u></p> <p>In accordance with applicable SRs</p> <p style="text-align: center;"><u>ED</u></p> <p style="text-align: center;"><u>B</u></p> <p style="text-align: center;"><u>3.8-17</u></p>

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air, and Turbocharger Air Assist

3.8-18

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air, and turbocharger air assist subsystems shall be within limits for each required diesel generator (DG). The fuel level for the stored diesel fuel oil shall be within the following limits:

PS

a. Combined storage of $\geq 65,000$ gal for two units in MODES 1, 2, 3, and 4, or

b. Combined storage of

3.8-10

1. $\geq 33,000$ gal for one unit (if any) in MODES 1, 2, 3, and 4, and

2. $\geq 26,000$ gal for each unit in MODES 5 and 6.

NOTE

The performance of diesel fuel oil tank cleaning requires one fuel oil storage tank to be removed from service to be drained and cleaned. During this time, the fuel oil storage requirement for one unit operation in MODES 1, 2, 3, and 4 and one unit operation in MODE 6 with at least 23 feet of water above the reactor vessel flange or with the reactor vessel defueled is $\geq 35,000$ gallons. The tank being cleaned may be inoperable for up to 10 days. For the duration of the tank cleaning, temporary onsite fuel oil storage of $\geq 24,000$ gallons will be maintained. Prior to removal of a tank from service, the offsite circuits required by LCO 3.8.1 or 3.8.2 will be verified to be OPERABLE.

3.8-11

APPLICABILITY: When associated DG(S) is required to be OPERABLE.

PS

ACTIONS

-----NOTE-----
 Separate Condition entry is allowed for each DG or fuel oil storage tank, except for Condition A.

3.8-10

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. Combined One or more DGs with fuel level < [33,000] gal and > [28,285] gal in storage tanks not within limits.</p>	<p>A.1.1 Verify combined fuel oil level = 29,000 gallons for each unit operating in MODES 1, 2, 3, or 4 AND A.1.2 Verify combined fuel oil level = 23,000 gallons for each unit operating in MODES 5 or 6 AND A.2 Restore fuel oil level to within limits.</p>	<p>Immediately <u>3.8-10</u></p> <p>Immediately</p> <p>48 hours</p>
<p>B. One or more DGsBoth units in MODE 1, 2, 3, or 4 with lube oil inventory < 650 gal and > 610 gal. OR One unit in MODE 1, 2, 3, or 4 and one unit in MODE 5 or 6 with lube oil inventory < 590 gal and > 520 gal.</p>	<p>B.1 Restore lube oil inventory to within limits.</p>	<p>48 hours <u>B-PS</u> <u>3.8-47</u></p>
<p>C. One or more fuel oil storage tanks DGs with stored fuel oil total particulates not within limit.</p>	<p>C.1 Restore fuel oil total particulates within limit.</p>	<p>7 days <u>3.8-10</u></p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. One or more fuel oil storage tanks DGs with new fuel oil properties not within limits.</p>	<p>D.1 Restore stored fuel oil properties to within limits</p>	<p>30 days <u>3.8-10</u></p>
<p>E. One or more DGs with both starting air receiver pressures < 180 [225] psig and ≥ 150 [125] psig.</p>	<p>E.1 Restore one starting air receiver pressure per DG to ≥ 180 [225] psig.</p>	<p>48 hours <u>3.8-12</u></p>
<p>F. One or more DGs with turbocharger air assist air receiver pressure < 180 psig and ≥ 150 psig.</p>	<p>F.1 Restore turbocharger air assist air receiver pressure to ≥ 180 psig.</p>	<p>48 hours <u>3.8-18</u></p>
<p>GF. Required Action and associated Completion Time not met.</p> <p><u>OR</u></p> <p>One or more DG's DGs diesel fuel oil, lube oil, turbocharger air assist or starting air subsystem not within limits for reasons other than Condition A, B, C, D, or E or F.</p>	<p>GF.1 Declare associated DG inoperable.</p>	<p>Immediately</p> <p><u>3.8-10</u></p> <p><u>3.8-18</u></p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H Required Action and associated Completion Time of Condition A, B, C, or D not met. OR Fuel oil storage tanks or lube oil not within limits for reasons other than Conditions A, B, C, or D.	H1 Declare all DGs on associated unit(s) inoperable. AND, if associated unit is in MODES 1, 2, 3, or 4.	Immediately 3.8-10
	H2 Be in MODE 3	6 hours
	AND H3 Be in MODE 5	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.3.1 Verify each fuel oil storage tank contains combined storage \geq [33,000] gal of fuel within limits.	31 days 3.8-10 B-PS (continued)
SR 3.8.3.2 Verify lubricating oil inventory is \geq [500] 650 gal.	31 days B-PS
SR 3.8.3.3 Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.3.4	Verify each DG has at least one air start receiver with a pressure is ≥ 180 [225] psig.	31 days <u>3.8-12</u> <u>B-PS</u>
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	31 days <u>B</u>
SR 3.8.3.6	Verify each DG turbocharger air assist air receiver pressure is ≥ 180 psig.	31 days <u>3.8-18</u>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.3.6 For each fuel oil storage tank:	10 years
a. Drain the fuel oil;	<u>3.8-31</u>
b. Remove the sediment; and	
c. Clean the tank.	

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 ~~Three Class 1E The Train A and Train B DC electrical power subsystems shall be OPERABLE.~~

3.8-13

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours
B. More than one full capacity charger receiving power simultaneously from a single 480 V vital bus OR Any DC bus not receiving power from its associated AC electrical power distribution subsystem.	B.1 Restore the DC electrical power subsystem to a configuration wherein each charger is powered from its associated 480 volt vital bus	14 days <u>3.8-13</u>
C.B. Required Action and Associated Completion Time not met.	C.B.1 Be in MODE 3. AND C.B.2 Be in MODE 5.	6 hours <u>3.8-13</u> 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.4.1 Verify battery terminal voltage is ≥ 130 [129] V on float charge.</p>	<p>7 [31] day S</p> <p><u>B-PS</u></p> <p><u>3.8-34</u></p>
(continued)	
<p>SR 3.8.4.2 Verify no visible corrosion at battery terminals and connectors.</p> <p>OR</p> <p>Verify battery connection resistance [is $\leq 150 \times 10^{-6}$ ohm \leq [1E-5 ohm] $\leq 150 \times 10^{-6}$ ohm for inter-cell connections, \leq [1E-5 ohm] $\leq 150 \times 10^{-6}$ ohm for inter-rack connections, \leq [1E-5 ohm] for inter tier connections, and $\leq 150 \times 10^{-6}$ ohm \leq [1E-5 ohm] for terminal connections].</p>	<p>92 days</p> <p><u>B-PS</u></p>
<p>SR 3.8.4.3 Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.</p>	<p>18 [12] months</p> <p><u>B-PS</u></p> <p><u>3.8-41</u></p>
<p>SR 3.8.4.4 Remove visible terminal corrosion, verify battery cell to cell and terminal connections are clean and tight, and are coated with anti-corrosion material.</p>	<p>18 [12] months</p> <p><u>B-PS</u></p>
<p>SR 3.8.4.5 Verify battery connection resistance is $\leq 150 \times 10^{-6}$ ohm \leq [1E-5 ohm] for inter-cell connections, \leq [1E-5 ohm] $\leq 150 \times 10^{-6}$ ohm for inter-rack connections, \leq [1E-5 ohm] for inter tier connections, and $\leq 150 \times 10^{-6}$ ohm for terminal connections].</p>	<p>18 [12] months</p> <p><u>B-PS</u></p> <p>(continued)</p>

SURVEILLANCE REQUIREMENT (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.4.6</p> <p style="text-align: center;"><u>NOTE</u></p> <p>This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.</p> <hr/> <p>Verify each battery charger supplies \geq 400 amps at \geq 130 [125] V for \geq 1 [8] hours.</p>	<p>3.8-39</p> <p>B</p> <p>3.8-20</p> <p>18 months</p> <p>B-PS</p>
<p>SR 3.8.4.7</p> <p style="text-align: center;"><u>NOTES</u></p> <p>1. The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7 once per 60 months.</p> <p>2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.</p> <hr/> <p>Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.</p>	<p>3.8-38</p> <p>3.8-20</p> <p>18 months</p> <p>B</p>

(continued)

SURVEILLANCE REQUIREMENT (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.4.8</p> <p>-----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR.</p> <p>-----</p> <p>Verify battery capacity is $\geq 80\%$ of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.</p>	<p><u>3.8-20</u></p> <p>60 months <u>B</u></p> <p>AND</p> <p>18 12 months when battery shows degradation or has reached 85% of expected life for the application with capacity < 100% of manufacturer's rating</p> <p>AND <u>3.8-16</u></p> <p>24 months when battery has reached 85% of the expected life with capacity $\geq 100\%$ of manufacturer's rating</p>

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5

~~The Class 1E~~ DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown."

3.8-13

APPLICABILITY:

MODES 5 and 6,

During movement of irradiated fuel assemblies.

ACTIONS

NOTE
LCO 3.0.3 is not applicable.

3.8-35

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more required DC electrical power subsystems inoperable.</p>	<p>A.1-1 Declare affected required feature(s) inoperable.</p>	<p>Immediately <u>ED</u></p>
	<p><u>OR</u></p>	
	<p>A.2.1 Suspend CORE ALTERATIONS.</p>	<p>Immediately</p>
	<p><u>AND</u></p>	
	<p>A.2.2 Suspend movement of irradiated fuel assemblies.</p>	<p>Immediately</p>
	<p><u>AND</u></p>	
	<p>A.2.3 Initiate action to suspend operations involving positive reactivity additions.</p>	<p>Immediately</p>
	<p><u>AND</u></p>	<p>(continued)</p>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY									
<p>SR 3.8.5.1 -----NOTE----- The following SRs are not required to be performed: SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. -----</p> <p>For DC sources required to be OPERABLE, the following SRs are applicable:</p> <table data-bbox="413 1092 1073 1218"> <tr> <td>SR 3.8.4.1</td> <td>SR 3.8.4.4</td> <td>SR 3.8.4.7</td> </tr> <tr> <td>SR 3.8.4.2</td> <td>SR 3.8.4.5</td> <td>SR 3.8.4.8.</td> </tr> <tr> <td>SR 3.8.4.3</td> <td>SR 3.8.4.6</td> <td></td> </tr> </table>	SR 3.8.4.1	SR 3.8.4.4	SR 3.8.4.7	SR 3.8.4.2	SR 3.8.4.5	SR 3.8.4.8.	SR 3.8.4.3	SR 3.8.4.6		<p>In accordance with applicable SRs</p>
SR 3.8.4.1	SR 3.8.4.4	SR 3.8.4.7								
SR 3.8.4.2	SR 3.8.4.5	SR 3.8.4.8.								
SR 3.8.4.3	SR 3.8.4.6									

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LC0 3.8.6 Battery cell parameters for ~~the three Class 1E Train A and Train B~~ batteries shall be within the limits of Table 3.8.6-1.

3.8-13

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more batteries with one or more battery cell parameters not within Category A or B limits.</p>	<p>A.1 Verify pilot cell[s] electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.</p> <p><u>AND</u></p>	<p>1 hour <u>B-PS</u></p>
	<p>A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits.</p> <p><u>AND</u></p>	<p>24 hours</p> <p><u>AND</u></p> <p>Once per 7 days thereafter</p>
	<p><u>AND</u></p> <p>A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.</p>	<p>31 days</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Required Action and associated Completion Time of Condition A not met.</p> <p><u>OR</u></p> <p>One or more batteries with average electrolyte temperature of the representative cells 60°F.</p> <p><u>OR</u></p> <p>One or more batteries with one or more battery cell parameters not within Category C values.</p>	<p>B.1 Declare associated battery inoperable.</p>	<p>Immediately</p> <p style="text-align: center;"><u>B</u></p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.6.1 Verify battery cell parameters meet Table 3.8.6-1 Category A limits.</p>	<p>7 ^{3.8-34} days</p> <p>(continued)</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.6.2 Verify battery cell parameters meet Table 3.8.6-1 Category B limits.</p>	<p>92 days 3.8-14</p> <p>AND</p> <p>Once within 7 days 24 hours after a battery discharge < 118 110 V</p> <p>AND B-PS</p> <p>Once within 7 days 24 hours after a battery overcharge > 145 150 V</p> <p>3.8-14</p> <p>B-PS</p>
<p>SR 3.8.6.3 Verify average electrolyte temperature of representative cells is $\geq 60^{\circ}\text{F}$.</p>	<p>92 days B</p>

SURVEILLANCE REQUIREMENTS (continued)

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameters Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and $\leq \frac{1}{4}$ inch above maximum level indication mark(a)	> Minimum level indication mark, and $\leq \frac{1}{4}$ inch above maximum level indication mark(a)	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity(b)(c)	≥ 1.195 [1.200]	≥ 1.190 [1.195] AND Average of all connected cells > 1.200 [1.205]	Not more than 0.020 below average of all connected cells AND Average of all connected cells ≥ 1.190 [1.195]

- (a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.
- (b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging current is < 2 amps when on float charge. B
ED
- (c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance. B

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 3.8-15
Four ~~Class 1E Vital 120 V UPS~~ Train A and Train B inverters shall be OPERABLE.

NOTE
[One/two] inverter[s] may be disconnected from [its/their] associated DC bus for < 24 hours to perform an equalizing charge on [its/their] associated [common] battery, provided:
a. The associated AC vital bus(es) [is/are] energized from [its/their] [Class 1E constant voltage source transformers] [inverter using internal AC source]; and
b. All other AC vital buses are energized from their associated OPERABLE inverters.

B-PS

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 -----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any vital 120 V AC bus de-energized. ----- Restore inverter to OPERABLE status.	<u>B</u> <u>3.8-15</u> 24 hours (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.7.1 Verify correct inverter voltage, [frequency.] and alignment to required AC vital buses.	7 days <u>B-PS</u>

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 The ~~Class 1E UPS~~ Inverters shall be OPERABLE to support the onsite Class 1E 120 VAC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown."

3.8-15

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

NOTE
LCO 3.0.3 is not applicable.

3:8-35

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately <u>B</u>
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
AND		
A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately (continued)	
AND		

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.8.1 Verify correct inverter voltage, [frequency.] and alignments to required AC vital buses.	7 days <u>B-PS</u>

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 ~~The required Class 1E Train A and Train B AC, DC, and 120 VAC vital bus electrical power distribution subsystems shall be OPERABLE.~~

3.8-15

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC electrical power distribution subsystems inoperable.	A.1 Restore AC electrical power distribution subsystem to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
B. One 120 VAC vital bus subsystem inoperable.	B.1 Restore 120 VAC vital bus subsystem to OPERABLE status.	2 hours <u>ED</u> <u>AND</u> <u>3.8-15</u> 16 hours from discovery of failure to meet LCO
C. One DC electrical power distribution subsystem inoperable.	C.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours <u>AND</u> 16 hours from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u> D.2 Be in MODE 5.	36 hours
E. Two trains required Class 1E AC, DC, or 120 VAC vital buses with inoperable distribution subsystems that result in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately <u>3.8-15</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.9.1 Verify correct breaker alignments and voltage to required AC, DC, and 120 VAC vital bus electrical power distribution subsystems.	7 days <u>B</u>

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portion of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE. 3.8-15

APPLICABILITY: MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTIONS

NOTE:
LCO 3.0.3 is not applicable.

3.8-35

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or 120 VAC vital bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately <u>3.8-15</u>
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	(Continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate actions to restore required AC, DC, and 120 VAC vital bus electrical power distribution subsystems to OPERABLE status.	Immediately <u>3:8-15</u>
	<p style="text-align: center;"><u>AND</u></p> A.2.5 Declare associated required residual heat removal subsystem(s) inoperable and not in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.10.1 Verify correct breaker alignments and voltage to required AC, DC, and 120 VAC vital bus electrical power distribution subsystems.	7 days <u>3:8-15</u>

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions -** The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions -** The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications -** The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes-** Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts -** The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

Methodology For Mark-up of NUREG-1431 Specifications
(Continued)

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES CONTENTS

Mark-up:

<u>BASES</u>	<u>PAGE</u>
3.8.1	B 3.8-1
3.8.2	B 3.8-68
3.8.3	B 3.8-79
3.8.4	B 3.8-97
3.8.5	B 3.8-116
3.8.6	B 3.8-122
3.8.7	B 3.8-135
3.8.8	B 3.8-142
3.8.9	B 3.8-149
3.8.10	B 3.8-168
Methodology	(1 Page)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of offsite power sources (~~preferred power sources,~~ normal and alternate(s)), and the onsite standby power sources (~~Train A and Train B three~~ diesel generators (DGs) ~~for each unit~~). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E AC Distribution System ~~for each unit~~ is divided into ~~redundant three~~ load groups (~~trains~~) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each ~~load group train~~ has connections to two preferred offsite power sources and a single DG.

Offsite power is supplied to the ~~230 kV and 500 kV unit~~ switchyard[s] from the transmission network by ~~two 230 kV transmission lines and three 500 kV transmission lines~~. From the switchyard[s], these two electrically and physically separated circuits provide AC power, through ~~step down station auxiliary and standby startup transformers~~, to the 4.16 kV ESF busses. A detailed description of the offsite power network and the circuits to the Class 1E buses is found in the FSAR, Chapter ~~8~~ (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite class 1E bus(es).

Certain required unit loads are returned to service in a predetermined sequence in order to

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System. Within [1] minute after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the load sequencer timers (auto transfer timers). Each individual timer connects a single ESF component.

The onsite standby power source for each 4.16 kV ESF bus is a dedicated DG. For Unit 1, DGs [11] and [12] 1-1, 1-2, and 1-3 are dedicated to ESF buses H, G, and F [11] and [12], respectively. For Unit 2, DGs 2-1, 2-2, and 2-3 are dedicated to ESF buses G, H, and F. A DG starts

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND
(continued)

automatically on a safety injection (SI) signal (i.e., e.g., low pressurizer pressure or high containment pressure signals), undervoltage on the offsite standby startup source, or on an ESF bus degraded voltage or undervoltage signal (refer to LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation"). After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with an SI signal. The DGS will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone. Following the trip of offsite power, a sequencer/an undervoltage signal strips nonpermanent loads from the ESF bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the automatic load sequencing timers (ESF timers) sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application. Each ESF component is provided with its own load sequencing timer.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within 1 minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for the six DGs Train A and Train B DGs satisfy the requirements of Regulatory Guide 1.9 (Ref.3). The continuous service rating of each DG is 2600 [7000] kW with [10] % overload

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

permissible for up to 2 hours in any 24 hour period. The ESF loads that are powered from the 4.16 kV ESF buses are listed in Reference 2.

Fuel oil is transferred from the storage tanks via the diesel fuel oil storage and transfer system to replenish the day tanks as required. The design incorporates sufficient redundancy so that a malfunction of either an active or a passive component will not impair the ability of the system to supply fuel oil. Two redundant fuel oil transfer pumps supply fuel oil to DG day tanks from either storage tank. One pump is adequate to supply the six DGs operating at full load. Each DG tank has two separate, redundant transfer pump start-stop level switches. Each level switch automatically starts a transfer pump and opens the supply header solenoid valve corresponding to the respective transfer pump, 0-1 or 0-2.

APPLICABLE
SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the FSAR, Chapter {6} (Ref. 4) and Chapter {15} (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the Accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of ~~ESF systems powered by~~ the onsite or offsite AC sources OPERABLE during Accident conditions in the event of:

- a. An assumed loss of all offsite power or all onsite AC power; and
- b. A worst case single failure.

The AC sources satisfy Criterion 3 of ~~10 CFR 50.36(c)(2)(iii)~~ NRC Policy Statement.

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power System and separate and independent DGs for each ~~train Class 1E ESF bus~~ ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit.

~~[In addition, one required automatic load sequencer per train must be OPERABLE.]~~

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

~~Offsite circuit #1 consists of Safeguards Transformer B, which is supplied from Switchyard Bus B, and is fed through breaker 52-3 powering the ESF transformer XNB01, which, in turn, powers the #1 ESF bus through its normal feeder breaker. Offsite circuit #2 consists of the Startup Transformer, which is normally fed from the Switchyard Bus A, and is fed through breaker PA 0201, powering the ESF transformer, which, in turn, powers the #2 ESF bus through its normal feeder breaker.~~

The Unit 1 Offsite Circuit #1 consists of Startup Transformer 1-1 supplied from the immediate access 230 KV Switchyard power source which feeds Startup Transformer 1-2 through series supply breakers 52VU12 and 52VU14. Startup Transformer 1-2 then supplies power through breaker 52HG15 to each vital bus feeder breaker (Bus F - 52HF14, Bus G - 52HG14, Bus H - 52HH14). The Unit 1 Offsite Circuit #2 is the delayed access 500 KV circuit which becomes available only after opening the motor operated disconnect to the main generator. This circuit consists of Auxiliary Transformer 1-2 supplied from the 500 KV Switchyard through the main bank transformers. Auxiliary Transformer 1-2 supplies power directly to each of the vital bus feeder breakers (Bus F - 52HF13, Bus G - 52HG13, Bus H - 52HH13).

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

The Unit 2 Offsite Circuit #1 consists of Startup Transformer 2-1 supplied from the immediate access 230 kV Switchyard power source which feeds Startup Transformer 2-2 through series supply breakers 52VU23 and 52VU24. Startup Transformer 2-2 then supplies power through breaker 52HG15 to each vital bus feeder breaker (Bus F - 52HF14, Bus G - 52HG14, Bus H - 52HH14). The Unit 2 Offsite Circuit #2 is a delayed access circuit which only becomes available after opening the motor operated disconnect to the main generator. This circuit consists of Auxiliary Transformer 2-2 supplied from the 500 kV Switchyard through the main bank transformers. Auxiliary Transformer 2-2 supplies power directly to each of the vital bus feeder breakers (Bus F - 52HF13, Bus G - 52HG13, Bus H - 52HH13).

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. ~~This will be accomplished within [10] seconds.~~ Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby with the engine ~~pre-lubed and pre-warmed at ambient conditions.~~ Additional DG capabilities must be demonstrated to meet required Surveillance, e.g., capability of the DG to ~~revert to standby status automatically~~ sequence the emergency loads onto the DG following opening of the ~~auxiliary breaker~~ on an EGCS ESF actuation signal while operating in parallel test mode.

~~Proper sequencing of loads, [including tripping of nonessential loads,] is a required function for DG OPERABILITY.~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

The AC sources ~~in one train~~ must be separate and independent (to the extent possible) ~~of the AC sources in the other train~~. For the DGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical. A circuit ~~is normally~~ may be connected to more than one ESF bus, with ~~OPERABLE~~ fast transfer capability to the other circuit ~~OPERABLE~~, and does not violate separation criteria. ~~A circuit that is not connected to an ESF bus is required to have OPERABLE fast transfer interlock mechanisms to at least two ESF buses to support OPERABILITY of that circuit.~~

~~The two redundant diesel fuel oil supply trains supply fuel oil to DG day tanks from either storage tank. One pump (supply train) is adequate to supply the six DGs operating at full load.~~

APPLICABILITY

The AC sources ~~[and sequencers]~~ are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design, limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

APPLICABILITY
(continued)

- b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources - Shutdown"

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered. ~~If one required offsite circuit to only one Class 1E 4160 VAC bus is inoperable, SR 3.8.1.1 needs to be performed only for the affected Class 1E 4160 VAC bus. It is not necessary to perform SR 3.8.1.1 for the other Class 1E 4160 VAC buses.~~

~~The 230 KV system should be considered inoperable when the DCPD Shift Supervisor has been notified of system inoperability by the Diablo Canyon Switching Center Grid Operations Scheduling, or Grid Shift Supervisor, in accordance with Transmission Operating Procedure O-23 "Operating Instructions for Reliable Transmission Service for Diablo Canyon P.P."~~

~~Reviewer's Note: The turbine auxiliary feedwater pump is only required to be considered a redundant required feature, and, therefore, required to be determined OPERABLE by this Required Action, if the design is such that the remaining OPERABLE motor or turbine driven~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

~~auxiliary feedwater pump(s) is not by itself capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.~~

A.2

~~Not used~~

~~Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features.~~

~~These features are powered from the redundant AC electrical power train. This includes motor driven auxiliary feedwater pumps. Single train systems, such as turbine driven auxiliary feedwater pumps, may not included.~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

ACTIONS

A.2 (continued)

~~Not used~~

~~The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:~~

- ~~a. The train has no offsite power supplying its loads; and~~
- ~~b. A required feature on the other train is inoperable.~~

~~If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.~~

~~Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.~~

~~The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AG~~

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

~~sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.~~

A.2

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

ACTIONS

A.2 (continued)

potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to ~~72 hours~~ ~~7 days~~. This could lead to a total of ~~144 hours~~ ~~10 days~~, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional ~~72 hours~~ ~~7 days~~ (for a total of ~~9~~ ~~17~~ days) allowed prior to complete restoration of the LCO. The ~~6~~ ~~10~~ day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between 72 hour and ~~6~~ ~~10~~ day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

~~As in Required Action A.2,~~ The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

"time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

ACTIONS

B.1 (continued)

the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

~~Reviewer's Note: The turbine driven auxiliary feedwater pump is only required to be considered a redundant required feature, and, therefore, required to be determined OPERABLE by this Required Action, if the design is such that the remaining OPERABLE motor or turbine driven auxiliary feedwater pump(s) is not by itself capable (without any reliance on the motor driven auxiliary feedwater pump powered by the emergency bus associated with the inoperable diesel generator) of providing 100% of the auxiliary feedwater flow assumed in the safety analysis.~~

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. ~~These features are powered from the three AC electrical power distribution subsystems (buses). Examples of required features would include, but are not limited to, auxiliary saltwater pumps, centrifugal charging pumps, or motor driven auxiliary feedwater pumps. A complete list of equipment encompassed by Required Action B.2 is provided by Reference 13. These features are designed with redundant safety related trains. This includes motor driven auxiliary feedwater pumps. Single train systems, such as turbine driven auxiliary feedwater pumps, are not included. Required features are redundant safety-related systems.~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

subsystems, trains, components, and devices that depend on the diesel generators as a source of emergency power. Since the turbine driven auxiliary feedwater pump is a single train system, it is not included (e.g., the turbine driven auxiliary feedwater pump and one motor driven auxiliary feedwater pump or two motor driven auxiliary feedwater pumps). Redundant required feature failures consist of inoperable features associated with one of the other Class 1E AC electrical power distribution subsystems a train, redundant to the train subsystem associated with that has an inoperable DG. An example: if DG 1-1 (Bus H) were declared inoperable with safety injection pump 1-1 (Bus F) already inoperable, SIP 1-2 (Bus H) would then be required to be declared inoperable within 4 hours, and TS 3.0.3 entered. A Note has been added to point out that during operation in Modes 1, 2, and 3, two auxiliary feedwater pumps are required to meet the redundant features requirement to mitigate a feedwater line break. If both of the available AFW pumps are motor driven, neither of the two may be supplied by the DG which is inoperable. For example, declaring DG 1-1 inoperable would require maintaining the turbine driven AFW pump and motor driven AFW pump 1-2 OPERABLE, while declaring DG 1-2 inoperable would only require any 2 of the 3 AFW pumps OPERABLE.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

ACTIONS

B.2 (continued).

- a. An inoperable DG exists; and
- b. A required feature, redundant to a required feature associated with the inoperable DG on

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

~~one of the other Class 1E AC electrical power distribution subsystems~~
~~the other train (Train A or Train B) is inoperable.~~

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable ~~redundant~~ required support or supported features, ~~or both,~~ that are associated with one of the OPERABLE DGs, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is Acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining ~~two~~ OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DGs, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DGs, the other DGs would be declared

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG[s]. performance of

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

ACTIONS

B.3.1 and B.3.2 (continued)

SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that those DGs. If a DG has already started and loaded on a bus, it is not necessary to shutdown the DG and perform SR 3.8.1.2. The DG is verified OPERABLE since it is performing its intended function.

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the {plant corrective action program} will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), {24} hours is reasonable to confirm that the OPERABLE DG{s} is are not affected by the same problem as the inoperable DG.

B.4

~~According to Regulatory Code 1.93 (Ref. 6), Operation may continue in Condition B for a period that should not exceed 72 hours 7 days. This AOT was revised from 72 hours to 7 days by License Amendment (LA) 44 for Unit 1 and LA 43 for Unit 2.~~

In Condition B, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 72-hour 7-day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of ~~144 hours~~ 10 days, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 9 13 days) allowed prior to complete restoration of the LCO. The 6 10 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

ACTIONS

B.4 (continued)

and B are entered concurrently. The "AND" connector between the 72-hour 7 day and 6 10 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. ~~The Completion Time for this failure of redundant required features is reduced to 12 hours, from that allowed for one train without offsite power (Required Action A.2).~~ The rationale for the reduction to 12 hours ~~for Required Action C.1~~ is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the valid case, and a shorter Completion Time of 12 hours is appropriate. ~~Required features are redundant safety-related systems, subsystems, trains, components, and devices that depend on the DGs as a source of emergency power. These features are powered from the three Class 1E AC electrical power distribution subsystems redundant AC safety trains. Examples of required features would include, but are not limited to, auxiliary saltwater pumps, centrifugal charging pumps, or motor-driven~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

~~auxiliary feedwater pumps. Since the turbine driven auxiliary feedwater pump is a single train system, it is not included. This includes motor driven auxiliary feedwater pumps. Single train features, such as turbine driven auxiliary pumps, are not included in the list.~~

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

- a. All required offsite circuits are inoperable; and
- b. A required feature is inoperable.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

ACTIONS

C.1 and C.2 (continued)

If at any time during the existence of Condition C (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

- a. The configuration of the redundant ~~Class IE~~ AC electrical power system that remains available is not susceptible to a single bus or switching failure; and
- b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA transient. In fact, a simultaneous loss of offsite AC sources, a ~~DBA~~ LOCA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

ACTIONS

C.1 and C.2 (continued)

continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

D.1 and D.2

~~Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition D are modified by a Note to indicate that when Condition D is entered with no AC source to any train, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems Operating," must be immediately entered. This allows Condition D to provide requirements for the loss of one offsite circuit and one DG, without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.~~

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 12 hours.

In Condition D, individual redundancy is lost in both the offsite electrical power system and ~~may be lost in~~ the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

for repairs, and the low probability of a DBA occurring during this period.

E.1

With Train A and Train B two or more DGs inoperable, there are no remaining standby AC sources the remaining onsite AC sources are inadequate. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

ACTIONS

E.1 (continued)

are available to power the minimum required ESF functions. Since the offsite electrical power system may be the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Reference 6, with both two or more DGS inoperable, operation may continue for a period that should not exceed 2 hours.

F.1

Condition F corresponds to a level of degradation in which one train of the DFO transfer system is inoperable. The onsite AC electrical power systems are redundant and available to support ESF loads. However, one subsystem required for the onsite AC electrical system operability has lost its redundancy (DFO supply to the DGs).

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The sequencer(s) is an essential support system to [both the offsite circuit and the DG associated with a given ESF bus]. [Furthermore, the sequencer is on the primary success path for

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

~~most major AC electrically powered safety systems powered from the associated ESF bus.] Therefore, loss of an [ESF bus sequencer] affects every major ESF system in the [division]. The [12] hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer OPERABILITY. This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods when the sequencer is inoperable is minimal.~~

~~This Condition is preceded by Note that allows the Condition to be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads under any conditions. Implicit in this Note is the concept that the Condition must be retained if any sequencer failure mode results in the inability to start all or part of the safety loads when required, regardless of power availability, or results in overloading the offsite power circuit to a safety bus during an event and thereby causes its failure. Also~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

ACTIONS

F.1 (continued)

~~implicit in the Note, is that the Condition is not applicable to any train that does not have a sequencer.~~

G.1

With both trains of DFO inoperable, the onsite AC sources are inadequate (loss of DFO supply to all DGs). With an assumed loss of offsite electrical power, insufficient AC sources are available to power the minimal required ESF functions. Since the offsite electrical power system is the only source for AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

HG.1 and HG.2

If the inoperable AC electric power sources cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

H I.1

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

Condition H corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system of the remaining offsite circuit will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

I

Condition I corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, further loss of a remaining DG will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE
REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGS are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 9), and Regulatory Guide 1.137 (Ref. 10), as addressed in the FSAR.

SURVEILLANCE
REQUIREMENTS

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

(continued)

state output voltage of [3740] 3785 V is 90% of the nominal 4160 V output voltage. This value, which is specified in ANSI C84.1 (Ref. 11), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating, consistent with the second level undervoltage relay allowable values. This is the minimum steady state voltage needed on the 4160 volt vital buses to ensure adequate 4160 volt, 480 volt and 120 volt levels. The specified maximum steady state output voltage or [4756] 4400 V is equal to the maximum operating voltage specified for 4000 V motors for 4000 V motors specified in ANSI C84.1 (Ref. 11). It The maximum steady state output voltage of 4400 V ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to $\pm 2\%$ of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2 for SR 3.8.1.2) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the DGS are started from standby conditions. Standby conditions

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

SURVEILLANCE
REQUIREMENTSSR 3.8.1.2 and SR 3.8.1.7 (continued)

for a DG means that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations. For the purposes of this SR, diesel generator start will be initiated using one of the following signals: 1) manual, 2) simulates loss of offsite power, or 3) safety injection actuation test signal.

In order to reduce stress and wear on diesel engines, some manufacturers recommend a modified start in which the starting speed of DGS is limited, warmup is limited to this lower speed, and the DGS are gradually accelerated to synchronous speed prior to loading. These start procedures are the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer. Currently, the DGS are not able to gradually accelerate, and Note 3 does not apply. However, if the DG's governor is replaced with a governor with the ability to allow gradual acceleration, Note 3 may be applied.

SR 3.8.1.7 requires that, at a 184 day Frequency, the DG starts from standby conditions and achieves required speed within 10 seconds and required voltage and frequency within 10 13 seconds. The 10 second start requirement reflects the point during the DG's acceleration at which the DG is assumed to be able to accept load. The 13 second start requirement reflects the point at which the DG is assumed to have reached stable operation. The 10 and 13 second start requirements supports the assumptions of the design basis LOCA analysis in the FSAR, Chapter [15] (Ref. 5).

The 10 and 13 second start requirements are ~~is~~ not applicable to SR 3.8.1.2 (see Note 3) when a modified start procedure as described above is used. If a modified start is not used, the 10 and 13 second start requirements of SR 3.8.1.7 apply applies.

Since SR 3.8.1.7 requires a 10 timed second start, it is more restrictive than SR 3.8.1.2.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.

The normal 31 day Frequency for SR 3.8.1.2 (see Table 3.8.1.1, "Diesel Generator Test Schedule," in the accompanying LCO) is consistent with Regulatory Guide 1.9 (Ref. 3). ~~is consistent with Generic Letter 94-01 (Ref. 12)~~ The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.1.3 (continued)

60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between {0.8 lagging} and {1.0}. The {0.8} value is the design rating of the machine, while the {1.0} is an operational limitation {to ensure circulating currents are minimized}. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The 31 day Frequency for this Surveillance (~~Table 3.8.1-1~~) is consistent with Regulatory Guide 1.9 (Ref. 3).

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time ~~per unit~~ in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

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SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank [and engine mounted tank] is at or above the level at which fuel oil is automatically added, is a contained quantity sufficient for DG operation at full load for a nominal one-hour period. One hour is adequate time for an operator to take corrective action to restore the fuel oil supply to the affected day tank. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour of DG operation at full load plus 10%.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since low level alarms are since the transfer pumps auto-starts are at a level above the minimum contained volume. Therefore, normal DG operation will not result in day tank level below the minimum required volume. Additional assurance of sufficient day tank contained volume is provided by a low level alarm.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASESSURVEILLANCE
REQUIREMENTSSR 3.8.1.4 (continued)

~~provided and facility operators would be aware of any large uses of fuel oil during this period.~~

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day ~~[and engine mounted]~~ tanks once every ~~[31]~~ days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 10). This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from ~~its associated the fuel oil~~ storage tanks to ~~its associated~~ each day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

The Frequency for this SR is variable, depending on individual system design, with up to a [92] day interval. The [92] day Frequency corresponds to the testing requirements for pumps as contained in the ASME Code.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASESSURVEILLANCE
REQUIREMENTS3.8.1.6 (continued)

~~Section XI (Ref. 11); however, the design of fuel transfer systems is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day [and engine mounted] tanks during or following DG testing. In such a case, a 31 day Frequency is appropriate. Since proper operation of fuel transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs.~~

The Frequency of 31 days is adequate to verify proper operation of the fuel oil transfer pumps and day tank supply valves to maintain the required volume of fuel oil in the day tanks. The frequency has been proven acceptable through operating experience.

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Transfer of each [4.16 kV ESF bus] power supply from the normal offsite circuit to the alternate offsite circuit, which is the immediate access 230 kV, demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. Transfer of each 4.16 kV ESF bus power supply from the alternate offsite circuit (immediate access 230 kV) to the delayed access circuit (500 kV circuit) demonstrates the ability of the delayed access circuit. The [18 month] Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

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the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. ~~Credit may be taken for unplanned events that satisfy this SR~~
~~This Note does not prohibit the application of LCO 3.0.5.~~

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASESSURVEILLANCE
REQUIREMENTSSR 3.8.1.9 (continued)

overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip: ~~[For this unit, the single load for each DG and its horsepower rating is as follows:]~~ The single largest DG load is a centrifugal charging pump (CCP), which is rated at 600 hp. The CCP has a maximum demand, based on the maximum expected horsepower input and motor efficiency, of 515 kW. This Surveillance may be accomplished by:

- a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or
- b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

~~As required by IEEE 308 (Ref. 12), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.~~

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The ~~3~~ 2.4 seconds specified is equal to 60% of a typical ~~5~~ 4 second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

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and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover following load rejection. The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9).

This SR is modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. ~~Credit may be taken for unplanned events that satisfy this SR. This Note does not prohibit the application of LCO 3.0.5.~~

In order to ensure that the DG is tested under load

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASESSURVEILLANCE
REQUIREMENTSSR 3.8.1.9 (continued)

conditions that are as close to design basis conditions as possible. Note 2 requires that, if synchronized to offsite power, testing must be performed using a power factor \leq [0.9] lagging. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

~~Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

- ~~a. Performance of the SR will not render any safety system or component inoperable;~~
- ~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and~~
- ~~c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.~~

SR 3.8.1.10

This Surveillance demonstrates the DG's capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG would experience following a full load rejection and verifies that the DG does not

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

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trip upon loss of the load. These acceptance criteria provide for DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

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In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor $\leq [0.9] 0.87$ lagging. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9) and is intended to be consistent with expected fuel cycle lengths.

This SR has been modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbation to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. ~~Credit may be taken for unplanned events that satisfy this SR~~
~~This Note does not prohibit the application of LCO 3.0.5.~~

~~Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

- ~~a. Performance of the SR will not render any safety system or component inoperable;~~
- ~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and~~
- ~~c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.~~

(Continued)

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B 3.8.1 AC Sources - Operating

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SR 3.8.1.11

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

SURVEILLANCE
REQUIREMENTSSR 3.8.1.11 (continued)

encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG autostart time of [10] seconds is derived from requirements of the accident analysis to respond to a design basis large-break LOCA accident. The 10 second requirement reflects the assumption of the accident analysis that the DG has reached the point in its acceleration where the DG is able to accept load. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability is achieved. After energization of the loads, steady state voltage and frequency are required to be within their limits.

The requirement to verify the connection and power supply of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. The permanently connected loads are the Class 1E 480 VAC buses. The permanently connected loads do not receive a load shed signal. In addition, the containment fan cooler units do not receive a load shed signal but are de-energized when their motor contactors drop out on undervoltage. The permanently connected loads are re-energized when the DG breaker closes to energize the bus. The auto-connected loads are those loads that are energized via their respective sequencing timer. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow.

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

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~~or residual heat removal (RHR) systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation.~~
In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG systems to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGS during testing. For the purpose of this testing, the DGS must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASESSURVEILLANCE
REQUIREMENTSSR 3.8.1.11 (continued)

consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. ~~This Note does not prohibit the application of LCO 3.0.5.~~

SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves stability by reaching the required voltage and frequency within the specified time (~~10~~ 13 seconds) from the design-basis Safety Injection actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.12.d and SR 3.8.1.12.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF Safety Injection signal without loss of offsite power. ~~The emergency loads are the ESF loads.~~

The requirement to verify the connection of permanent and autoconnected loads to the immediate access 230 kV offsite power system is intended to satisfactorily show the relationship of these loads to the DG loading logic. ~~For a description of the permanent and auto-connected loads, see SR 3.8.1.11 Bases.~~ In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. ~~For instance, ECCS injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation.~~ In lieu of actual demonstration of connection and loading of loads, testing that adequately shows

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

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the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of {18 months} takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the {18 month} Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASESSURVEILLANCE
REQUIREMENTSSR 3.8.1.12 (continued)

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGS during testing. For the purpose of this testing, the DGS must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. ~~Credit may be taken for unplanned events that satisfy this SR. This Note does not prohibit the application of LCO 3.0.5.~~

SR 3.8.1.13

This Surveillance demonstrates that DG noncritical protective functions (e.g., high jacket water temperature) are bypassed on a loss of voltage signal concurrent with an ESF actuation test signal. ~~When the diesel engine trip cutout switch is in the cutout position and the DG is aligned for automatic operation and critical protective functions (engine overspeed, generator differential current, and low lube oil pressure, high crankcase pressure, and start failure relay) trip the DG to avert substantial damage to the DG unit. The noncritical trips include directional power loss of field, breaker overcurrent, high jacket water temperature, and diesel overcrank. These noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

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The [18 month] Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service. ~~Credit may be taken for unplanned events that satisfy this SR~~ **THIS NOTE** does not prohibit the application of LCO 3.0.5.

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

~~Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

- ~~a. Performance of the SR will not render any safety system or component inoperable;~~
- ~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and~~
- ~~c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.~~

SR 3.8.1.14

Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3), requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ~~≥ [2] hours~~ of which is at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing must be performed using a power factor of ~~≤ [0.9]~~ ~~0.87~~ lagging. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

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The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

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REQUIREMENTS

SR 3.8.1.14 (continued)

The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. ~~Credit may be taken for unplanned events that satisfy this SR~~ ~~This Note does not prohibit the application of LCO 3.0.5.~~

SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve ~~stability by reaching~~ the required voltage and frequency within [10] 13 seconds. The [10] 13 second time is derived from the requirements of the accident analysis to respond to a design basis ~~large break LOCA accident~~. The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(5).

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

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recommendations in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least [2] hours at full load conditions prior to performance of this Surveillance is based on test data and manufacturer recommendations, which indicate 45 minutes is sufficient for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASESSURVEILLANCE
REQUIREMENTS
(continued)SR 3.8.1.16

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and the DG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open and can receive an auto close signal on bus undervoltage, and the load sequencing timers are reset.

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. ~~This Note does not prohibit the application of LCO 3.0.5.~~

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing. ~~A Safety Injection signal, received while the DG is operating in a test mode, results in the auxiliary breaker opening and the emergency loads automatically sequencing onto the DG, and the DG will automatically reset to ready to load operation if a LOCA actuation signal is received during operation in the test~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

~~mode. Ready to load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE 308 (Ref. 13), paragraph 6.2.6(2).~~

~~The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading was not affected by the DG operation in test mode. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable.~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.1.17 (continued)

This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(8), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. ~~Credit may be taken for unplanned events that satisfy this SR. This Note does not prohibit the application of LCD 3.0.5.~~

SR 3.8.1.18

Under accident [and loss of offsite power] conditions, loads are sequentially connected to the bus by the [automatic load sequencer] load sequencer timers. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The [10] load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

~~With an ESF timer found to be outside the range of acceptable settings, the corresponding DG shall be declared inoperable in MODES 1, 2, 3, and 4, and the corresponding CONDITION followed. With an Auto Transfer timer found to be outside~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

~~the range of acceptable settings, the corresponding DG shall be declared inoperable for all MODES.~~

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. ~~This Note does not prohibit the application of LCO 3.0.5.~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.1.18 (continued)

~~Reviewer's Note: The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:~~

- ~~a. Performance of the SR will not render any safety system or component inoperable;~~
- ~~b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems; and~~
- ~~c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.~~

SR 3.8.1.19

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with an ~~ESF actuation~~ ~~Safety Injection~~ signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

The Frequency of {18 months} takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of {18 months}.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGS during testing. For

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.1.19 (continued)

the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGS. The reason for Note 2 is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. ~~This Note does not prohibit the application of LCO 3.0.5.~~

SR 3.8.1.20

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGS are started simultaneously.

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9).

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGS must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

Diesel Generator Test Schedule

~~The DG test schedule (Table 3.8.1.1) implements the recommendations of Revision 3 to Regulatory Guide 1.9 (Ref. 3). The purpose of this test schedule is to provide timely test data to establish a confidence level associated with~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

~~the goal to maintain DG reliability > 0.95 per demand.~~

~~According to Regulatory Guide 1.9, Revision 3 (Ref. 3), each DG should be tested at least once every 31 days. Whenever a DG has experienced 4 or more valid failures in the last 25 valid tests, the maximum time between tests is reduced to 7 days. Four failures in 25 valid tests is a failure rate of 0.16, or the threshold of acceptable DG performance, and~~

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

SURVEILLANCE
REQUIREMENTS

~~Diesel Generator Test Schedule (continued)~~

~~hence may be an early indication of the degradation of DG reliability. When considered in the light of a long history of tests, however, 4 failures in the last 25 valid tests may only be a statistically probable distribution of random events. Increasing the test Frequency will allow for a more timely accumulation of additional test data upon which to base judgment of the reliability of the DG. The increased test Frequency must be maintained until seven consecutive, failure free tests have been performed.~~

~~The Frequency for accelerated testing is 7 days, but no less than 24 hours. Tests conducted at intervals of less than 24 hours may be credited for compliance with Required Actions. However, for the purpose of re-establishing the normal 31-day Frequency, a successful test at an interval of less than 24 hours should be considered an invalid test and not count towards the 7 consecutive failure free starts, and the consecutive test count is not reset.~~

A test interval in excess of 7 days (or 31 days, as appropriate) constitutes a failure to meet the SRs, and results in the associated DG being declared inoperable. It does not, however, constitute a valid test or failure of the DG, and any consecutive test count is not reset.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter 8.
3. Regulatory Guide 1.9, Rev. 3, [date] July 1993.
4. FSAR, Chapter 6.
5. FSAR, Chapter 15.

(Continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability." July 2, 1984.
8. 10 CFR 50, Appendix A, GDC 18.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

REFERENCES
(continued)

9. Regulatory Guide 1.108, Rev. 1, August 1977.
 10. Regulatory Guide 1.137, Rev. 1, Oct 1979.
 11. ASME, Boiler and Pressure Vessel Code, Section XI.
 12. ~~IEEE Standard 308-1978~~ Generic Letter 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators." May 31, 1994
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating"

APPLICABLE
SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

within the Required Actions. This allowance is in recognition that certain testing and maintenance activities) must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.
- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.
- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of ~~10 CFR 50.36(c)(2)(ii)~~ the NRC Policy Statement.

BASES

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with the Class 1E AC electrical power distribution subsystem train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to

BASES

LCO
(continued)

provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es). Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit.

~~Offsite circuit #1 consists of Safeguards Transformer B, which is supplied from Switchyard Bus B, and is fed through breaker 52-3 powering the ESF transformer XNB01, which, in turn, powers the #1 ESF bus through its normal feeder breaker. The second offsite circuit consists of the Startup Transformer, which is normally fed from the Switchyard Bus A, and is fed through breaker PA 0201 powering the ESF transformer, which, in turn, powers the #2 ESF bus through its normal feeder breaker.~~

The Unit 1 Offsite Circuit #1 consists of Startup Transformer 1-1 supplied from the immediate access 230 kV Switchyard power source, which feeds Startup Transformer 1-2 through series supply breakers 52VU12 and 52VU14. Startup Transformer 1-2 then supplies power through breaker 52HG15 to each vital bus feeder breaker (Bus F - 52HF14, Bus G - 52HG14, Bus H - 52HH14). The Unit 1 Offsite Circuit #2 is the delayed access 500 kV circuit which becomes available only after opening the motor operated disconnect to the main generator. This circuit consists of Auxiliary Transformer 1-2 supplied from the 500 kV Switchyard through the main bank transformers. Auxiliary Transformer 1-2 supplies power directly to each of the vital bus feeder breakers (Bus F - 52HF13, Bus G - 52HG13, Bus H - 52HH13).

BASES

The Unit 2 Offsite Circuit #1 consists of Startup Transformer 2-1 supplied from the immediate access 230 kV Switchyard power source, which feeds Startup Transformer 2-2 through series supply breakers 52VU23 and 52VU24. Startup Transformer 2-2 then supplies power through breaker 52HG15 to each vital bus feeder breaker (Bus F - 52HF14, Bus G - 52HG14, Bus H - 52HH14). The Unit 2 Offsite Circuit #2 is a delayed access circuit which only becomes available after opening the motor operated disconnect to the main generator. This circuit consists of Auxiliary Transformer 2-2 supplied from the 500 kV Switchyard through the main bank transformers. Auxiliary Transformer 2-2 supplies power directly to each of the vital bus feeder breakers (Bus F - 52HF13, Bus G - 52HG13, Bus H - 52HH13).

The DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. ~~This sequence must be accomplished within [10] seconds.~~ The DG must be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. ~~These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby at ambient conditions.~~

BASES

LCO
(continued)

~~Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.~~

~~In addition, proper sequencer operation is an integral part of offsite circuit OPERABILITY since its inoperability impacts on the ability to start and maintain energized loads required OPERABLE by LCO 3.8.10.~~

With administrative controls in place, it is acceptable for Class 1E AC electrical power distribution subsystems trains to be cross tied during shutdown conditions, allowing a single offsite power circuit or a single DG to supply all the required Class 1E AC electrical power distribution subsystems trains.

The two redundant diesel fuel oil transfer pumps supply fuel oil to DG day tanks from either storage tank. One pump is adequate to supply the six DGs operating at full load. Only one train is required to be OPERABLE in MODES 5 or 6.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

- a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;
- b. Systems needed to mitigate a fuel handling accident are available;
- c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

BASES

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8..

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1

An offsite circuit would be considered inoperable if it were not available to one the required ESF train Class IE bus(es). Although two trains are required by LCO 3.8.10, the one train with two Class IE AC electrical power distribution subsystems are required by LCO 3.8.10, and one Class IE AC electrical power distribution subsystem has offsite power available, the remaining Class IE AC electrical power distribution subsystem may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By the allowance of allowing the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

ACTIONS
(continued)

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required AC electrical power distribution subsystems trains, the option would still exist

BASES

to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SDM is maintained.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not an AC electrical power distribution subsystem train is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a

BASES

de-energized AC electrical power distribution subsystem train.

BASES

SURVEILLANCE
REQUIREMENTSSR 3.8.2.1

SR 3.8.2.1 requires ~~lists~~ the SRs from LCO 3.8.1 that are necessary applicable for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.17 is not required to be met because the required OPERABLE DG(s) is not required to undergo periods of being synchronized to the offsite circuit. SR 3.8.1.20 is excepted because starting independence is not required with the DG(s) that is not required to be operable. SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.18 (for ESF timers) and SR 3.8.1.19 are excepted because SI response functions are not required to be operable.

This SR is modified by a Note listing applicable SRs from LCO 3.8.1 that are not required to be performed. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of an SR, and to ~~The note would also preclude deenergizing a required 4160 V ESF bus or disconnecting a required offsite circuit during for performance of an SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG. The note does not except the requirement for the DG, 4160 V ESF bus, or offsite circuit to be capable of performing the particular function, just that the capability of need not be demonstrated while that source of power is being relied on to support meeting the LCO. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE.~~

Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES

None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air, and Turbocharger Air Assist

BASES

BACKGROUND

~~Each diesel generator (DG) is provided with a storage tank having a fuel oil capacity sufficient to operate that diesel for a period of 7 days while the DG is supplying maximum post loss of coolant accident load demand discussed in the FSAR, Section [9.5.4.2] (Ref. 1). The maximum load demand is calculated using the assumption that a minimum of any two DGs is available. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time to replenish the onsite supply from outside sources.~~

~~Fuel oil is transferred from storage tank to day tank by either of two transfer pumps associated with each storage tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve or tank to result in the loss of more than one DG. All outside tanks, pumps, and piping are located underground.~~

~~The diesel fuel oil storage system consists of two common tanks with a nominal capacity of 40,000 gallons each. The IS-required fuel oil quantity is based on the calculated fuel oil consumption necessary to support the operation of the DGs to power the minimum engineered safety feature (ESF) systems required to mitigate a design basis accident (LOCA) in one unit and those minimum required systems for a concurrent non-LOCA safe shutdown in the remaining unit (both units initially in Mode 1 operation). The fuel oil consumption is calculated for a period of 7 days operation of minimum ESF systems. This requirement provides a sufficient operating period within which offsite power can be restored and/or additional fuel can be delivered to the site.~~

~~Fuel oil is transferred from the storage tanks via the diesel fuel oil storage and transfer~~

BASES

system to replenish the day tanks as required. The design incorporates sufficient redundancy so that a malfunction of either an active or a passive component will not impair the ability of the system to supply fuel oil. Two redundant fuel oil transfer pumps supply fuel oil to DG day tanks from either storage tank. One pump is adequate to supply the six DGs operating at full load. Each DG tank has two separate, redundant transfer pump start-stop level switches. Each level switch automatically starts a transfer pump and opens the supply header solenoid valve corresponding to the respective transfer pump, 0-1 or 0-2. In addition, high and low level alarms are provided on each day tank and activate alarms both locally and in the control room.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. Each engine oil sump contains an inventory capable of supporting a minimum of [7] days of operation. [The onsite storage in addition to the engine oil sump is sufficient to ensure 7 days of continuous operation.] The diesel lube oil storage requirement is based upon a conservative usage factor of 1% of fuel oil consumption. The storage system used to meet this requirement is that located in the warehouse where 650 gallons of lube oil is stored in drums. This storage is augmented by a second storage location within the diesel engine itself. The lube oil level on each engine's dip stick is maintained 5 inches above the engine's operability limit. This provides approximately 120 gallons of usable lube oil within each of the 6 diesel engines. This supply is sufficient to allow the operator to replenish lube oil from outside sources.

BASES

APPLICABLE
SAFETY ANALYSES

Each DG has an two redundant 100% capacity air start systems and a turbocharger air assist system with adequate capacity for five three successive start attempts each on the DG without recharging the air start receiver(s) or the turbocharger air assist air receiver. The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 4), and in the FSAR, Chapter [15] (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, and the air start, and turbocharger air assist subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) the NRC Policy Statement.

LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of full-load minimum ESF systems operation. The required combined stored diesel fuel oil is a contained quantity with different storage requirements for unit operation in MODE 1, 2, 3, and 4 and for MODE 5 and 6. With both units operating in MODE 1, 2, 3, and 4, the required level is $\geq 65,000$ gallons. With one unit operating in MODE 1, 2, 3, or 4, and the other unit in MODE 5 or 6, the required fuel oil level is 33,000 gallons plus 26,000 gallons, for a total of 59,000 gallons combined storage. With both units in MODE 5 or 6, the required fuel oil level is 52,000 gallons. The required combined stored fuel oil

(Continued)

BASES

was revised by License Amendment 74 for Unit 1 and 73 for Unit 2.

The Note permits diesel fuel oil storage tank cleaning to be performed. Each tank is required to be cleaned on a 10-year frequency. Conducting the cleaning requires the tank to be taken out of service. For this infrequent event, the inventory in the remaining tank is sufficient to support operation of the DGs to power the minimum required loads to maintain safe conditions for a period of 4 days, considering one unit in MODE 1, 2, 3, 4, 5, or 6 and one unit in MODE 6 with 23 feet of water above the reactor vessel flange or with the reactor vessel dewatered. The requirements for diesel fuel oil tank cleaning were approved by License Amendment 74 for Unit 1 and 73 for Unit 2.

The fuel oil is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate at full load for 7 days. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the

(Continued)

BASES

LCO
(continued)

availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (A00) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown."

The starting air system and turbocharger air assist system are ~~is~~ required to have a minimum capacity for ~~five~~ ~~three~~ successive DG start attempts without recharging the air start receivers ~~or the turbocharger air assist air receiver.~~

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an A00 or a postulated DBA. Since stored diesel fuel oil, lube oil, and the starting air, and turbocharger air assist subsystems support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil, and starting air, and turbocharger air assist are required to be within limits when the associated DGs ~~is~~ are required to be OPERABL.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG ~~or diesel fuel oil storage tank, except for Condition A.~~ This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition

(Continued)

BASES

entry and application of associated Required Actions.

Condition A is excepted from this allowance for diesel fuel oil storage tanks. Since the requirement is for a combined storage quantity contained in both storage tanks. However, the Note would still allow separate Condition entry into a DG subsystem's Required Action coincident with Condition A.

A.1 and A.2

In this Condition, the 7 day fuel oil supply for a the DGs is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6 day supply. These circumstances may be caused by events, such as full load operation required after an inadvertent start while at minimum required level, or feed and bleed operations, which may be necessitated by increasing particulate levels or any number of other oil quality degradations. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the associated DGs inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period. Should the specified 6 day fuel oil supply for both units not be available, but the available supply is still greater than that required to support operation of one unit, then that available supply can be allocated to a selected unit, and the DGs declared inoperable under Action H need

(Continued)

BASES

ACTIONS
(continued)

only be the ones associated with the unit that has the inadequate supply.

B.1

With diesel engine lube oil stored inventory < 500 650 gal, sufficient lubricating oil to support 7 days of continuous DG operation at full load conditions may not be available based on minimum 7 day ESF systems loading at 1% of fuel oil consumption. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply of 610 gallons. This ACTION should be entered based upon warehouse inventory of less than 650 gallons with both units in MODES 1, 2, 3 or 4 and less than 590 gallons with one unit in MODES 1, 2, 3, or 4 and the other in MODES 5 or 6. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.5.3. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is

(Continued)

BASES

unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG's inoperable. The 7 day Completion Time allows for further evaluation, re-sampling and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.4 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

(Continued)

BASES

ACTIONS
(continued)

E.1

With both starting air receiver pressures < 180 [225] psig, sufficient capacity for five [three] successive DG start attempts does not exist. However, as long as the one receiver pressure is > 150 [125] psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the one air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

F.1

With turbocharger air assist air receiver pressure < 180 psig, sufficient capacity for three successive DG start attempts does not exist. However, as long as the receiver pressure is > 150 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the turbo air assist air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

GE.1

With a Required Action and associated Completion Time not met, or one or more DG's fuel oil, lube oil, or starting air, or turbocharger air assist subsystem not within limits for reasons other than addressed by Conditions A through D B, E

(Continued)

BASES

or F, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

H1, H2, and H3

With a Required Action and associated Completion Time not met, or the fuel oil storage tanks not within limits for reasons other than addressed by Conditions A, C, or D, the fuel oil storage tanks may be incapable of supporting the DGs in performing their intended function. This condition requires declaring inoperable, all the DGs on the unit(s) associated with either the inadequate fuel oil inventory, the fuel storage tank(s) having particulate outside the limit, and/or the fuel storage tank(s) having properties outside limits, and shutting down to MODE 3 in 6 hours and MODE 5 in 36 hours any associated unit(s) operating in MODE 1, 2, 3, or 4.

SURVEILLANCE
REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.3.1 (continued)

~~for 7 days at full load DG operation for 7 days based a realistic (minimum) ESF systems loading profile.~~ The 7 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.2

This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of full load operation for each DG at minimum ESF systems loading. The [500] 650 gal requirement is based on the DG manufacturer consumption values for the run time of the DG at 1% of fuel oil consumption. Implicit in this SR is the requirement to verify the capability to transfer the lube oil from its storage location to the DG, when the DG lube oil sump does not hold adequate inventory for 7 days of full load operation without the level reaching the manufacturer recommended minimum level. The storage system used to meet this requirement is that located within the warehouse where 650 gallon of lube oil is stored in drums.

A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the unit staff.

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated

(Continued)

BASES

with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days. The tests, limits, and applicable ASTM Standards are as follows:

- a. Sample the new fuel oil in accordance with ASTM D4057-~~81~~ (Ref. 6);
- b. Verify in accordance with the tests specified in ASTM D975-~~81~~ (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60°F of $\geq 27^\circ$ and $\leq 39^\circ$, a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point of $\geq 125^\circ\text{F}$; and
- c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-~~81~~ or a water and sediment content of ≤ 0.05 volume percent when tested in accordance with ASTM D-1796-83 (Ref. 6).

SURVEILLANCE
REQUIREMENTS

SR 3.8.3.3 (continued)

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

Within ~~31~~ 30 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-~~81~~ ~~81~~ (Ref. 7) are met for new fuel oil when tested in accordance with

(Continued)

BASES

ASTM D975-~~79~~ 81 (Ref. 6), except that the analysis for sulfur may be performed in accordance with ASTM D1552-~~79~~ 79 (Ref. 6) or ASTM D2622-~~79~~ 82 (Ref. 6). The ~~31~~ 30 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

~~If the analysis of the new fuel oil sample indicates that one or more of the other properties specified in Table 1 of ASTM D975-81 are not within limits, then Required Action D.1 shall be entered, allowing 30 days to restore fuel oil properties to within limits.~~

Fuel oil degradation during long term storage shows up as an increase in particulates, due mostly to oxidation. The presence of particulates does not mean the fuel oil will not burn properly in a diesel engine. The particulates can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-~~78~~ 78, Method A (Ref. 6). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. ~~For those designs in which the total stored fuel oil volume is contained in two or more interconnected tanks, each tank must be considered and tested separately.~~

~~ASTM D 2276-78 was written specifically for aviation fuel. However, it is used in this SR to evaluate diesel fuel oil. Therefore, it may be necessary to perform this test as a modified method. For example, a 500 ml sample may be analyzed rather than a one gallon sample.~~

(Continued)

BASES

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of ~~five~~ six engine start cycles without recharging. ~~[A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed.] Each start cycle is 15 seconds of cranking.~~ The pressure specified in this SR is intended to reflect the lowest value at which the ~~five~~ three starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every ~~31~~ days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, and contaminated fuel oil, and ~~or~~ from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water

(Continued)

BASES

does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

(Continued)

BASES

**SURVEILLANCE
REQUIREMENTS
(continued)**

SR 3.8.3.6

This Surveillance ensures that, without the aid of the refill compressor, sufficient turbocharger air assist air receiver capacity for each DG is available. The system design requirements provide for a minimum of six engine start cycles without recharging. Each start cycle is 15 seconds of cranking. The pressure specified in this SR is intended to reflect the lowest value at which three starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal turbocharger air assist air receiver pressure.

~~Draining of the fuel oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning are required at 10 year intervals by Regulatory Guide 1.137 (Ref. 2), paragraph 2.f. This SR also requires the performance of the ASME Code, Section XI (Ref. 8), examinations of the tanks. To preclude the introduction of surfactants in the fuel oil system, the cleaning should be accomplished using sodium hypochlorite solutions, or their equivalent, rather than soap or detergents. This SR is for preventive maintenance. The presence of sediment does not necessarily represent a failure of this SR, provided that accumulated sediment is removed during performance of the Surveillance~~

REFERENCES

1. FSAR, Section 9.5.4.2.
2. Regulatory Guide 1.137.
3. ANSI N195-1976, Appendix B.
4. FSAR, Chapter 6.

(Continued)

BASES

5. FSAR, Chapter 15.
 6. ASTM Standards: D4057-[] B1; D975-[] B1; D4176-[] B2; D1796
B3; D1552-[] B9; D2622-[] B2; D2276-78, Method A.
 7. ASTM Standards, D975, Table 1.
 8. ASME, Boiler and Presser Vessel Code, Section XI.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The station Class 1E DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred backup 120 VAC vital bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the Class 1E DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The [125/250] VDC electrical power system consists of two three independent and redundant safety related Class 1E DC electrical power subsystems ([Train A and Train B]). Each subsystem consists of [two] one 60-cell 125 VDC batteries [(each battery (Batteries 11(21), 12(22), and 13(23)) [50% capacity]), the associated dedicated battery charger(s) and backup charger for each battery, and all the associated switchgear control equipment, and interconnecting cabling.

The 250 VDC source is obtained by use of the two 125 VDC batteries connected in series. Additionally there is [one] spare battery charger per subsystem, which There are two backup chargers for the three Class 1E DC subsystems. One backup charger is shared between two Class 1E DC subsystems. The other backup charger is dedicated to the third Class 1E DC subsystem. The backup chargers provides backup service in the event that the preferred battery charger is out of service. If the spare backup battery charger is substituted for one of the preferred battery chargers, then the requirements of independence and redundancy between subsystems are not maintained, and

BASES

~~operation is this condition is limited to 14 days by Condition B.~~

During normal operation, the ~~[125/250]~~ VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

The ~~[Train A and Train B]~~ DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, ~~[4.16]~~ kV switchgear, and ~~[480]~~-V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn ~~are backup sources to~~ power the ~~120~~ VAC vital buses.

BASES

BACKGROUND
(continued)

The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution System - Operating," and LCO 3.8.10, "Distribution Systems - Shutdown."

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours and to perform three complete cycles of intermittent loads as discussed in the FSAR, Chapter [8] (Ref. 4).

Each 125 VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. ~~There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.~~

The batteries for ~~Train A and Train B~~ ~~the three~~ DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. ~~Battery size is based on 125% of required capacity and, after selection of an available commercial battery, results in a battery capacity in excess of 150% of required capacity.~~ The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery ~~discussed in the FSAR, Chapter [8] (Ref. 4).~~ The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

Each ~~Train A and Train B~~ DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 24 ~~12~~ hours while supplying normal

BASES

steady state loads discussed in the FSAR, Chapter [8] (Ref. 4).

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 6), and in the FSAR, Chapter 15 (Ref. 7), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite AC power or all onsite AC power; and
- b. A worst case single failure.

The DC sources satisfy Criterion 3 of ~~10 CFR 50.36(c)(2)(ii)~~ the NRC Policy Statement.

LCO

The DC electrical power subsystems, each subsystem consisting of ~~two~~ ~~one~~ ~~batteries~~ ~~battery~~, battery charger ~~for each battery~~ and the corresponding control equipment and interconnecting cabling supplying power to the associated bus ~~within the train~~ are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any ~~train~~ ~~one~~ DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires ~~all~~ ~~required~~ ~~the~~ ~~batteries~~ ~~battery~~ and respective ~~its~~ ~~normal~~ ~~or~~ ~~backup~~ chargers to be operating and connected to the associated DC bus(es).

BASES

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

BASES

APPLICABILITY
(continued)

- b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.5, "DC Sources - Shutdown."

ACTIONS

A.1

Condition A represents one train Class 1E DC electrical power subsystem and associated ESF equipment with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected train subsystem. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution subsystem train.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystems has have the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure would, however, result in the complete loss of the one of the two remaining 125 VDC electrical power subsystems with attendant loss of ESF functions, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 8) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

BASES

BC.1

The design of the 125 VAC electrical power distribution system is such that a battery can have associated with it a full capacity charger powered from its associated 480 VAC vital bus or an alternate full capacity charger powered from another 480 VAC vital bus. However, operation in the latter condition or with two chargers powered by the same vital bus is limited to 14 days.

BC.1 and BC.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5

BASES

ACTIONS

BC.1 and BC.2 (continued)

within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 8).

SURVEILLANCE
REQUIREMENTSSR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 7-31 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 9).

SR 3.8.4.2

Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each intercell, interrack, ~~intertier~~, and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The resistance of cell-to-cell connecting cables does not have to be included in measurement of connection resistance.

BASES

~~The limits established for this SR must be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.~~

The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of physical deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The 12 18 month Frequency for this SR is consistent with IEEE 450 (Ref. 9), which recommends detailed visual inspection of cell condition and rack integrity on a yearly basis, based on operational experience related to battery integrity and physical degradation.

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of intercell, interrack, ~~intertier~~, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.4. The resistance of cell-to-cell connecting cables does not have to be included in measurement of connection resistance for SR 3.8.4.5.

~~Reviewer's Note: The requirement to verify that terminal connections are clean and tight applies only to nickel cadmium batteries as per IEEE Standard P1106, "IEEE Recommended Practice for Installation, Maintenance, Testing and~~

BASES

~~Replacement of Vented Nickel-Cadmium Batteries for Stationary Applications." This requirement may be removed for lead acid batteries.~~

~~The connection resistance limits for SR 3.8.4.5 shall be no more than 20% above the resistance as measured during installation, or not above the ceiling value established by the manufacturer.~~

The Surveillance Frequencies of 12 ~~18~~ months is consistent with IEEE 450 (Ref. 9), which recommends cell to cell and terminal connection resistance measurement on a yearly basis, based on operational experience related to corrosion and connection resistance trends

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.8.4.6

This SR requires that each battery charger be capable of supplying ~~{400}~~ amps at and ~~{125}~~ ~~130~~ V for \geq ~~{8}~~ hours. These requirements are based on the design capacity of the chargers (Ref. 4). According to Regulatory Guide 1.32 (Ref. 10), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these ~~{18 month}~~ intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

~~This Surveillance is required to be performed during MODES 5 and 6 since it would require the DC electrical power subsystem to be inoperable during performance of the test.~~

~~This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR.~~

SR 3.8.4.7

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements ~~as specified in Reference 4.~~

BASES

The Surveillance Frequency of {18 months} is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 10) and Regulatory Guide 1.129 (Ref. 11), which state that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests, not to exceed {18 months}.

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.4.7 (continued)

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test ~~once per 60 months.~~

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the ~~one minute~~ duration of time equal to that of the service test.

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test. ~~The modified performance discharge test and service test should be performed in accordance with IEEE-450 (Ref. 9).~~

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. ~~Credit may be taken for unplanned events that satisfy this SR. This Note does not prohibit the application of LCO 3.0.5.~~

(Continued)

BASES

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

(Continued)

BASESSURVEILLANCE
REQUIREMENTSSR 3.8.4.8 (continued)

A battery modified performance discharge test is described in the Bases for SR 3.8.4.7. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 9) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected service life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 18 12 months. However, if the battery shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity > 100% of the manufacturer's rating. Degradation is indicated, according to IEEE-450 (Ref. 9), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is > [10%] below < 90% of the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE 450 (Ref. 9). The Surveillance Frequency basis is consistent with IEEE 450 (Ref. 9), except if accelerated testing is required, it will be performed at an 18-month frequency to coincide with a refueling outage.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system

(Continued)

BASES

and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR ~~305~~. This Note does not prohibit the application of LCO

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-1978.

(Continued)

BASES

REFERENCES
(continued)

4. FSAR, Chapter 8.
 5. IEEE-485-~~1983~~ 1978, June 1983.
 6. FSAR, Chapter 6.
 7. FSAR, Chapter 15.
 8. Regulatory Guide 1.93, December 1974.
 9. IEEE-450-~~1987~~ 1995.
 10. Regulatory Guide 1.32, February 1977.
 11. Regulatory Guide 1.129, December 1974.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The DC sources satisfy Criterion 3 of ~~10 CFR 50.36(c)(2)(i)~~ the NRC Policy Statement.

BASES

LCO

The DC electrical power subsystems, each subsystem consisting of ~~two~~ ~~batteries~~ ~~one battery~~, one battery charger per battery, and the corresponding control equipment and

BASES

LCO
(continued)

interconnecting class 1E cabling within the subsystem train, are required to be OPERABLE to support required trains of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown." An OPERABLE subsystem consists of a DC bus connected to a battery with an OPERABLE battery charger which is fed from an OPERABLE AC vital bus. The OPERABLE AC vital bus must have an OPERABLE DG capable of starting and automatic loading in the event of a LOOP. This ensures that the DC bus and battery will be powered by a battery charger in the event of a LOOP. With administrative controls in place, DC buses may be cross-tied when a battery is taken out for maintenance provided that the battery and the Class 1E cross-tie has sufficient capacity and protection for its own loads and the cross-tie loads. The resulting circuit is not required to be single failure resistant. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

- a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;
- b. Required features needed to mitigate a fuel handling accident are available;
- c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

BASES

- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

~~If two trains are required by LCO 3.8.10, the remaining train with DC power available may be one or more required DC electrical power subsystems may be inoperable provided that the remaining OPERABLE DC electrical power subsystem(s) support the DC electrical power distribution subsystem(s) required by LCO 3.8.10. "Distribution Systems Shutdown," and are capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel~~

BASES

ACTIONS
(continued).

movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the

allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

SURVEILLANCE
REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through

(Continued)

BASES

SR 3.8.4.8. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of an SRs. This note does not except the requirement for the battery to be capable of performing the particular function, just that the capability need not be demonstrated while that source of power is being relied on to meet the LCO. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. FSAR, Chapter 6.
 2. FSAR, Chapter 15.
-

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND

This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4, "DC Sources - Operating," and LCO 3.8.5, "DC Sources - Shutdown."

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least one train of the required DC sources electrical power subsystem(s) OPERABLE during accident conditions, in the event of:

- a. An assumed loss of all offsite AC power or all onsite AC power; and
- b. A worst case single failure.

Battery cell parameters satisfy the Criterion 3 of ~~10 CFR 50.36(c)(2)(11)~~ the NRC Policy Statement.

BASES

LCO

Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Electrolyte limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met.

BASES

APPLICABILITY

The battery cell parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery electrolyte OPERABILITY is only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

ACTIONS

A.1, A.2, and A.3

With one or more cells in one or more batteries not within limits (i.e., Category A limits not met, Category B limits not met, or Category A and B limits not met) but within the Category C limits specified in Table 3.8.6-1 in the accompanying LCO, the battery is degraded but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met and operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery cells. One hour provides time to ~~inspect~~ verify the electrolyte level and to confirm the float voltage of the pilot cells. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery is still capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because specific gravity measurements must be obtained for each connected cell. Taking into

(Continued)

BASES

consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A or B limits. This periodic verification is consistent with the normal Frequency of pilot cell Surveillances.

(Continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below less than 60°F, are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

SURVEILLANCE
REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters on a 31-day frequency are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of pilot cells.

(Continued)

BASES

SR 3.8.6.2

~~The quarterly inspection of battery cell specific gravity and voltage is consistent with IEEE 450 (Ref. 3). The quarterly inspection of specific gravity is more conservative than IEEE-450 (Ref. 3), which requires a yearly frequency. In addition, within 24 hours 7 days of a battery discharge < [110] 118 V or a battery overcharge > [150] 145 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to < [110] 118 V, do not constitute a battery discharge~~

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.6.2 (continued)

provided the battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells is $\leq 60^{\circ}\text{F}$, is consistent with a recommendation of IEEE-450 (Ref. 3), that states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on battery sizing calculations manufacturer recommendations.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for the each designated pilot cell in each battery. The cells selected as the pilot cells are those is that whose temperature, voltage, and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450

(Continued)

BASES

(Ref. 3), with the extra $\frac{1}{4}$ inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote a to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS

Table 3.8.6-1 (continued)

suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the recommendations of IEEE-450 (Ref. 3), which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells:

The Category A limit specified for specific gravity for each pilot cell is $\geq [1.200] [1.195]$ (0.015 below the manufacturer minimum fully charged ~~nominal~~ specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 3), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature and level. ~~For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The correction factors are provided by the battery manufacturer.~~ The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the minimum normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for

(Continued)

BASES

specific gravity for each connected cell is \geq ~~[1.195]~~ ~~1.190~~ (0.020 below the manufacturer ~~minimum~~ fully charged, ~~nominal~~ specific gravity) with the average of all connected cells $>$ ~~[1.205]~~ ~~1.200~~ (0.010 below the manufacturer ~~minimum~~ fully charged, ~~nominal~~ specific gravity). These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS

Table 3.8.6-1 (continued)

Category C defines the ~~minimum allowable~~ limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists, and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limits for float voltage is based on IEEE-450 (Ref. 3), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit of average specific gravity ≥ 1.195 ~~1.190~~ is based on manufacturer recommendations (0.020 below the manufacturer recommended ~~minimum~~ fully charged, ~~nominal~~ specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

The footnotes to Table 3.8.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is $< \{2\}$ amps on float charge. This current provides, in general, an indication of ~~a battery in a charged~~ overall battery condition.

(Continued)

BASES

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 3). Footnote (c) to Table 3:8.6-1 allows the float charge current to be used as an alternate to specific gravity for

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS

Table 3.8.6-1 (continued)

up to ~~7~~ days following a battery recharge. Within ~~7~~ days, each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than ~~7~~ days.

~~Reviewer's Note: The value of [2] amps used in footnote (b) and (c) is the nominal value for float current established by the battery vendor as representing a fully charged battery with an allowance for overall battery condition.~~

REFERENCES

1. FSAR, Chapter 6.
 2. FSAR, Chapter 15.
 3. IEEE-450-1980 ~~1995~~.
-

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND

The Class 1E UPS inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station battery provides an uninterruptible power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in the FSAR, Chapter 15 (Ref. 1).

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:

BASES

- a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and
- b. A worst case single failure.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of ~~10 CFR 50.36(c)(2)(ii)~~ the ~~NRC Policy Statement~~.

BASES

LCO

The ~~Class 1E UPS~~ inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The four inverters ~~[(two per train)]~~ ensure an uninterruptible supply of AC electrical power to the ~~120 VAC~~ vital buses even if the 4.16 kV safety buses are de-energized.

Operable inverters require the associated ~~120 VAC~~ vital bus to be powered by the inverter with output voltage and frequency within tolerances, and power input to the inverter from a ~~[125 VDC]~~ station battery. Alternatively, power supply may be from an internal AC source via rectifier as long as the station battery is available as the uninterruptible power supply.

~~This LCO is modified by a Note that allows [one/two] inverters to be disconnected from a [common] battery for < 24 hours, if the vital bus(es) is powered from a [Class 1E constant voltage transformer or inverter using internal AC source] during the period and all other inverters are operable. This allows an equalizing charge to be placed on one battery. If the inverters were not disconnected, the resulting voltage condition might damage the inverter[s]. These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 24 hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected AC vital bus while taking into consideration the time required to perform an equalizing charge on the battery bank.~~

~~The intent of this Note is to limit the number of inverters that may be disconnected. Only those inverters associated with the single~~

BASES

~~battery undergoing an equalizing charge may be disconnected. All other
inverters must be aligned to their associated batteries, regardless of
the number of inverters or unit design.~~

BASES

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."

ACTIONS

A.1

With a required inverter inoperable, its associated 120 VAC vital bus becomes inoperable until it is ~~manually~~ re-energized from its ~~Class 1E constant voltage source transformer or inverter using internal AC source~~.

For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures that the vital 120 VAC bus is re-energized within 2 hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the 120 VAC vital bus is powered from its

(Continued)

BASES

constant voltage source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the 120 VAC vital buses is the preferred source for powering instrumentation trip setpoint devices.

BASES

ACTIONS
(continued)

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. FSAR, Chapter ~~7~~[8].
 2. FSAR, Chapter 6.
 3. FSAR, Chapter 15.
-
-

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters-Shutdown

BASES

BACKGROUND A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The ~~DC to-AC Class 1E UPS~~ inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each ~~120~~ VAC vital bus during MODES 5 and 6 ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

BASES

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of ~~10 CFR 50.36(c)(2)(ii)~~ the ~~NRC Policy Statement~~.

BASES

LCO

The Class 1E UPS inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the 120 VAC vital buses even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters Class 1E 120 VAC vital bus requires that the 120 VAC vital bus be powered by the inverter. An OPERABLE Class 1E UPS inverter is one that is connected to an OPERABLE DC subsystem (see B 3.8.5). The resulting circuit is not required to be single failure resistant. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

- a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;
- b. Systems needed to mitigate a fuel handling accident are available;
- c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

(Continued)

BASES

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

If two trains are required by LCO 3.8.10, "Distribution Systems Shutdown," the remaining OPERABLE Inverters may be One or more Class 1E UPS inverters may be inoperable provided that the remaining OPERABLE inverters support the Class 1E 120 VAC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems Shutdown," and are capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare

(Continued)

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

required features inoperable with the associated Class 1E/UPS inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required Class 1E/UPS inverters and to continue this action until restoration is accomplished in order to provide the necessary Class 1E/UPS inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer.

SURVEILLANCE
REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification

(Continued)

BASES

of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the 120 VAC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

BASES

REFERENCES

1. FSAR, Chapter '6.
 2. FSAR, Chapter 15.
-
-

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems—Operating

BASES

BACKGROUND

~~The onsite Class 1E AC, DC, and AC vital bus electrical power distribution systems are divided by train into [two] redundant and independent AC, DC, and AC vital bus electrical power distribution subsystems.~~

The onsite Class 1E electrical power distribution system is designed with three 4160 V and 480 V vital buses (F, G, and H) and three 125 V DC vital buses. The plant protection system (PPS) is designed with four input channels (I, II, III, and IV) powered from four 120 VAC vital buses (1, 2, 3, and 4). The four channels provide input to the solid state protection system (SSPS) Trains A and B. Each SSPS train actuates engineered safety feature (ESF) equipment in the three vital AC and DC buses and certain non-vital equipment in the non-vital AC and DC buses.

~~There are three~~ The AC electrical power subsystems, each comprised of ~~for each train consists of a primary Engineered Safety Feature (ESF)~~ 4.16 kV bus and secondary [480 and 120] V buses, distribution panels, motor control centers and load centers. Each [4.16 kV ESF bus] has at least ~~one~~ two separate and independent offsite source of power as well as a dedicated onsite diesel generator (DG) source. Each [4.16 kV ESF bus] is normally connected to a preferred the 500 kV offsite source. After a loss of the preferred this normal 500kV offsite power source to a 4.16 kV ESF bus, a transfer to the alternate 230 kV offsite source is accomplished by utilizing a time delayed bus undervoltage relay. If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ESF bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries. Additional description of this system may be found in the Bases for

BASES

LCO 3.8.1, "AC Sources-Operating," and the Bases for LCO 3.8.4, "DC Sources-Operating."

The secondary 480 VAC electrical power distribution system for each train bus includes the safety related load centers, motor control centers, and distribution panels shown in Table B 3.8.9-1.

The 120 VAC vital buses are arranged in four buses two load groups per train and are normally powered from the inverters. The alternate power supply for the 120 VAC vital buses are Class 1E constant voltage source transformers powered from the same bus train as the associated inverter, and its use is governed by LCO 3.8.7, "Inverters-Operating." Each constant voltage source transformer is powered from a Class 1E AC bus. In addition, each inverter can be powered from a bus other than its associated bus.

There are two three independent 125/250 VDC electrical power distribution subsystems (one for each bus train).

The list of all required distribution buses is presented in Table B 3.8.9-1.

BASES

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 1), and in the FSAR, Chapter 15 (Ref. 2), assume ESF systems are OPERABLE. The Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite power or all onsite AC electrical power; and
- b. A worst case single failure.

The distribution systems satisfy Criterion 3 of ~~10 CFR 50.36(c)(2)(iii)~~ the NRC Policy Statement.

LCO

The required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of Class 1E AC, DC, and 120 VAC vital bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The Class 1E AC, DC, and 120 VAC vital bus

(Continued)

BASES

electrical power distribution subsystems are required to be OPERABLE.

Maintaining the ~~Train A and Train B Class 1E~~ AC, DC, and 120 VAC vital bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

(Continued)

BASES

LCO
(continued)

OPERABLE Class 1E AC electrical power distribution subsystems require the associated buses and ~~load centers, motor control centers, and distribution panels~~ to be energized to their proper voltages. OPERABLE Class 1E DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. OPERABLE 120 VAC vital bus electrical power distribution subsystems require the associated buses to be energized to their proper voltage from the associated ~~inverter via inverted DC voltage, inverter using internal AC source, or Class 1E constant voltage transformer~~.

In addition, tie breakers between redundant safety related Class 1E AC, DC, and 120 VAC vital bus power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, that could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 4.16 kV buses from being powered from the same offsite circuit.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

(Continued)

BASES

- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.10, "Distribution Systems - Shutdown."

(Continued)

BASES

ACTIONS

A.1

With one or more required Class 1E AC buses, load centers, or motor control centers, or distribution panels, except 120 VAC vital buses, in one train AC electrical power distribution electrical power subsystems inoperable and a loss of function has not yet occurred, the remaining portions of the AC electrical power distribution subsystems in the other train is are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining portions of the power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required Class 1E AC buses, load centers, and motor control centers, and distribution panels must be restored to OPERABLE status within 8 hours.

Condition A worst scenario is one train AC electrical power distribution subsystem without AC power (i.e., no offsite power to the 4160 V ESF bus train and the associated DG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining AC electrical power distribution subsystems train by stabilizing the unit, and on restoring power to the affected subsystem train. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

- a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected subsystem train, to the actions associated with taking the unit to shutdown within this time limit; and

(Continued)

BASES

- b. The potential for an event in conjunction with a single failure of a redundant component in the ~~other AC electrical power distribution subsystems~~ train with AC power.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 2 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again

(Continued)

BASES

ACTIONS

A.1 (continued)

become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With one or more 120 VAC vital bus subsystems inoperable and a loss of function has not yet occurred, the remaining OPERABLE 120 VAC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum [required] ESF functions not being supported. Therefore, the required AC vital bus subsystems must be restored to OPERABLE status powered from an alternate source within 2 hours by powering the bus from the associated inverter via inverted DC inverter using internal AC source or Class 1E constant voltage transformer. The required AC vital bus subsystems must then be re-powered by restoring its associated inverter to OPERABLE status within 24 hours under LCO 3.8.7 ACTION A.1.

Condition B represents one 120 VAC vital bus without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining

(Continued)

BASES

vital buses and restoring power to the affected ~~120 VAC~~ vital bus
~~subsystems~~.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate vital ~~120 VAC~~ power. Taking exception to LCO 3.0.2 for components without adequate vital ~~120 VAC~~ power, that would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

(Continued)

BASES

ACTIONS

B.1 (continued)

- a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate vital 120 VAC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected subsystem train; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the 120 VAC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE 120 VAC vital buses, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the 120 VAC vital bus distribution system. At this time, an AC bus train could again become inoperable, and 120 VAC vital bus distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed

(Continued)

BASES

outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

BASES

ACTIONS
(continued)

C.1

With one or more DC bus(es) in one train electrical power distribution subsystems inoperable and a loss of function has not yet occurred, the remaining portions of the DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining portion of the DC electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the [required] DC buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

Condition C represents one or more DC electrical power distribution subsystems train without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning for the affected bus(es). In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining DC electrical power distribution subsystems trains and restoring power to the affected subsystems train.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

- a. The potential for decreased safety by requiring a change in unit conditions

(Continued)

BASES

(i.e., requiring a shutdown) while allowing stable operations to continue;

- b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected ~~subsystem~~ train; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).

(Continued)

BASES

ACTIONS

C.1 (continued)

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC bus train could again become inoperable, and DC distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

(Continued)

BASES

With two trains Condition E corresponds to required Class 1E AC, DC, or 120 VAC vital buses with inoperable distribution subsystems that result in a loss of safety function, adequate core cooling, containment OPERABILITY and other vital functions for DBA mitigation would be compromised, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the [required] Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC, DC, and 120 VAC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. FSAR, Chapter 6.
 2. FSAR, Chapter 15.
 3. Regulatory Guide 1.93, December 1974.
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Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

LCO 3.8.9 CONDITION A
4160 VAC and 480 VAC

VOLTAGE	BUS F MAJOR ESF LOADS (TRAIN A)	BUS G MAJOR ESF LOADS (TRAIN B)	BUS H MAJOR ESF LOADS (TRAIN A&B)
4160 VAC	ASW PP 1 AFW PP 3 CCP PP 1 CCW PP 1 SI PP 1 480 VAC BUS F	ASW PP 2 CS PP 1 RHR PP 1 CC PP 2 CCW PP 2 480 VAC BUS G	AFW PP 2 (B) CS PP 2 (A) RHR PP 2 (A) SI PP 2 (B) CCW PP 3 (A&B) 480 VAV BUS H
480 VAC *	CFCU 1 CFCU 2	CFCU 3 CFCU 5	CFCU 4 (A&B)

* Partial listing of loads

LCO 3.8.9 CONDITION B
120 VAC

BUS 1 PY11 (21)** PY11A (21A)**	BUS 2 PY12 (22)**	BUS 3 PY13 (23)** PY13A (23A)**	BUS 4 PY14 (24)**
IY Powered by: 480 VAC BUS F/DC BUS 1 or TRY1 Powered by: 480 VAC BUS F or Backup 480 VAC BUS G	IY1 Powered by: 480 VAC BUS G/DC BUS 2 or TRY2 Powered by: 480 VAC BUS G or Backup 480 VAC BUS F	IY Powered by: 480 VAC BUS H/DC BUS 3 or TRY3 Powered by: 480 VAC BUS H or Backup 480 VAC BUS G	IY Powered by: 480 VAC BUS H/DC BUS 2 or TRY1 Powered by: 480 VAC BUS H or Backup 480 VAC BUS F

** Unit 2 in parentheses

LCO 3.8.9 CONDITION C
125 VDC

DC BUS 1 - Powered From:	DC BUS 2 - Powered From:	DC BUS 3 - Powered From:
Battery 1 and Battery Charger 11 (21)** or Battery Charger 121 (221)**	Battery 2 and Battery Charger 12 (22)** or Battery Charger 121 (221)**	Battery 3 and Battery Charger 131 (231)** or Battery Charger 132 (232)**

** Unit 2 in Parentheses

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND A description of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

BASES

- c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The Class 1E AC and DC and 120 VAC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(iii) the NRC Policy Statement.

BASES

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. An OPERABLE AC subsystem shall consist of a 4kV vital bus powered from at least one energized offsite power source with the capability of being powered from an OPERABLE DG. The DG may be the DG associated with that bus or, with administrative controls in place, a DG that can be cross-tied (via the startup cross-tie feeder breakers) to another bus. However, credit for this cross-tie capability cannot be taken credit for in those LCOs which specifically require an OPERABLE emergency power source. The latter ensures that the 4 kV bus will be immediately available after a LOOP without operator action. An OPERABLE DC subsystem consists of an OPERABLE DC bus (see B 3.8.5). An OPERABLE Class 1E 120 VAC subsystem consists of a vital 120 VAC bus that is powered by its OPERABLE inverter which is connected to an OPERABLE DC bus, or except as precluded by LCO 3.8.8, one that is powered from its associated vital 120 VAC regulating transformer that is selected to be powered from an OPERABLE AC vital bus. This ensures that the vital 120 VAC bus is capable of supplying either uninterruptable power from its associated inverter, or with administrative controls in place, from its vital 120 VAC regulating transformer after a brief time delay for the DG to load the bus following a LOOP. The 120 VAC regulating transformer must be capable of being energized without any operator action. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components—all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe

(Continued)

BASES

manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC and DC and 120 VAC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

- a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;
- b. Systems needed to mitigate a fuel handling accident are available;
- c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

The AC, DC, and 120 VAC vital bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

(Continued)

BASES

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, the movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant subsystems trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions).

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC, and 120 VAC electrical power distribution subsystems and to continue this action until

(Continued)

BASES

restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal (RHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the ~~Class 1E~~ AC, DC, and ~~120~~ VAC vital bus electrical power distribution subsystems are functioning properly, with all the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. FSAR, Chapter 6.
 2. FSAR, Chapter 15.
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Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(5 Pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.8

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER

JUSTIFICATION

- 3.8-01 NUREG-1431, LCO 3.8.1 assumes a two bus Class 1E vital bus, two DG AC electrical system configuration. Plant-specific system configuration is such that Unit 1 and Unit 2 each have three Class 1E vital buses and three DGs. ITS LCO 3.8.1b. and Conditions E. and H., have been revised and new Conditions I. and J. added to accommodate this difference, consistent with plant configuration, DCPD CTS, and the intent of the NUREG-1431 requirements.
- 3.8-02 A 10-day maximum ACTION statement entry meets the intent of the NUREG-1431 requirement, which is to provide a limit for the total time of not meeting the ITS LCO 3.8.1. NUREG-1431 lists a Completion Time of 6 days. This maximum AOT is equal to consecutive one DG inoperable and one A.C. Circuit inoperable (Conditions A. and B.) entries. Since the DCPD CTS includes a DG AOT of 7 days, rather than the NUREG-1431 assumed 72-hour AOT, the proposed maximum 10-day ITS AOT is 4 days longer than the NUREG-1431 limit of 6 days but still retains the intent.
- 3.8-03 In accordance with CTS, ITS LCO 3.8.1 Required Action B.4 Completion Time to restore DG to OPERABLE would be 7 days.
- 3.8-04 ITS LCO 3.8.1c, Conditions F, and G would be revised to separately address the condition of DFO transfer system inoperabilities. Proposed ITS Condition F is consistent with current licensing basis and is more restrictive than NUREG-1431. Proposed Condition G Completion Time is more restrictive than NUREG-1431 Condition E. for two DGs inoperable. The requirement to have at least one train of the DFO transfer system operable during CORE ALTERATIONS or movement of irradiated fuel assemblies is added to ITS 3.8.2 Condition B consistent with current licensing bases. These locations are revised and used since DCPD design does not include automatic load sequencers.
- 3.8-05 Consistent with TSTF-37, Rev.1, TS 5.6.7, "EDG Failure Report," Table 3.8.1-1, "Diesel Generator Test Schedule," and the associated changes to ITS SR 3.8.1.2 and SR 3.8.1.3 are revised to be consistent with the recommendations of NRC GL 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," dated May 31, 1994.
- GL 94-01 required that staff approval of these TS changes be contingent on a commitment to implement a maintenance program for monitoring and maintaining EDG performance in accordance with 10 CFR 50.65 and the guidance contained in RG 1.160. Since this program was required to be implemented by July 10, 1996, the requirements of GL 94-01 are currently being met.
- 3.8-06 ITS SR 3.8.1.8 and ITS SR 3.8.1.12 would be modified for plant system configuration and current licensing basis (CTS 4.8.1.1.1b.1) and CTS 4.8.1.1.1b.2)). Transfer of AC power from the normal offsite circuit to the alternate offsite circuit (automatically and manually) and to the delayed access circuit (manually) would be verified by ITS SR 3.8.1.8. ITS SR 3.8.1.12 would be modified to specify that the preferred, immediate access source is the part of the offsite system from which permanently connected and emergency loads would be energized on an actual or simulated SI signal.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.8

CHANGE NUMBER

JUSTIFICATION

- 3.8-07 This change is not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 3.8-08 ITS SR 3.8.1.17 would be revised to reflect the plant-specific design of the DGs for response to an actual or simulated SI signal while operating in test mode. Upon this condition, the SI signal would override the test mode by opening the auxiliary breaker and automatically sequencing the emergency loads onto the DG.
- 3.8-09 ITS SR 3.8.1.18 would be revised to reflect the plant-specific design of the load sequence timers. The NUREG-1431 requirement is worded for plants that are designed with an automatic load sequencer. The NUREG-1431 requirement would be revised to reflect plant-specific design, which utilizes individual load sequence timers for ESF and auto transfer.
- 3.8-10 ITS 3.8.3 would be revised to reflect plant-specific DFO storage system design and CTS. These changes meet the intent of the NUREG-1431 requirement.
- 3.8-11 To address the DCP CTS Conditions required by performance of DFO tank cleaning, a note would be added to ITS LCO 3.8.3. This Note corresponds to CTS 3.8.1.1 note. The Note added to LCO 3.8.3 would allow a reduced required fuel oil level during performance of DFO tank cleaning.
- 3.8-12 ITS LCO 3.8.3, Condition E., Required ACTION E.1., and SR 3.8.3.4 for the DG air start subsystem have been revised to reflect plant-specific air start system design and licensing basis. NUREG-1431 would be revised for this difference. [Each DG has two separate 100 percent capacity air start systems. Only 1 of the 2 air start receivers is necessary to provide the 3 DG starts required by DCP licensing basis.]
- 3.8-13 ITS LCO 3.8.4, LCO 3.8.5, and LCO 3.8.6 would be revised for DCP's plant configuration. At DCP, each nuclear unit is provided with 3 Class 1E DC electrical power subsystems consisting of the following major components: 3 60-cell, 125 V batteries; 3 separate 125 Vdc power distribution switchgear assemblies; and 5 battery chargers: NUREG-1431 is revised for these plant-specific differences.
- ITS LCO 3.8.4 Condition B. would be added and existing Action B re-labeled as Action C allowing 2 full capacity chargers to receive power simultaneously from a single 480 volt vital bus, or 1 DC bus to receive power from another AC division, provided the system is restored to a configuration wherein each charger is powered from its associated 480 volt vital bus within 14 days. The addition of Condition B to ITS LCO 3.8.4 is based on CTS 3.8.2.1 ACTION C. and is consistent with current DCP design and licensing basis. The design of the 125 volt DC distribution system is such that a battery can have associated with it a full capacity charger powered from its associated 480 volt vital bus or an alternate full capacity charger powered from another 480 volt vital bus.
- 3.8-14 ITS SR 3.8.6.2 is revised from 24 hours to be consistent with current licensing basis, which includes a frequency of once per 7 days. This frequency is adequate to meet the intent of the IEEE 450 recommendations regarding inspections following a severe discharge or overcharge.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.8

<u>CHANGE NUMBER</u>	<u>JUSTIFICATION</u>
3.8-15	ITS LCO 3.8.2, LCO 3.8.7, LCO 3.8.8, LCO 3.8.9, and LCO 3.8.10 are revised to reflect the DCPD design. DCPD's design provides: 3 AC, 3 DC, and 4 AC vital (with 4 inverters) bus electrical power distribution subsystems. Therefore, the NUREG-1431 LCO 3.8.2 would be revised to delete reference to "train" requirements. NUREG-1431, LCO 3.8.7, and LCO 3.8.9 wording, "Train A and Train B," does not apply and would also be replaced with applicable terminology.
3.8-16	ITS SR 3.8.4.8 would be revised to be consistent with current licensing basis, which requires an 18-month frequency for testing when the battery has seen degradation or has reached 85 percent of its service life for the manufacturer's rating for the application and was less than 100 percent capacity on the last performance test. NUREG-1431 allows 24-month surveillance interval when service life is beyond 85 percent but greater than or equal to 100 percent capacity on the last performance test would be retained. A description of degradation would also be included in ITS SR 3.8.4.8 Bases.
3.8-17	ITS SR 3.8.2.1 would be revised to list only applicable SRs and include as exceptions in the note those SRs that are applicable but should not be performed in MODE 5 or 6.
3.8-18	ITS LCO 3.8.3, Conditions, and SRs revised to include DG turbocharger air assist air receiver pressure requirements. Proper air pressure in the DG turbocharger air assist air receiver is necessary for the DG to reach voltage and frequency within the time assumed by safety analyses. However, adding the DG turbocharger air assist air receiver pressure requirements to ITS LCO 3.8.3 would address the situation where the support system is degraded but still capable of supporting the associated DG.
3.8-19	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
3.8-20	This change would eliminate the potential confusion that may arise, with respect to the application of an unplanned event which satisfies the requirements of a given SR, by including a discussion in the Bases of SR 3.0.1. The intent of the current notes throughout the ITS 3.8 SRs is applicable to any SR. The revision to the Bases for SR 3.0.1 will provide the necessary clarification so that the usage of this allowance can be applied consistently throughout the ITS. This change is consistent with TSTF-8.
3.8-21	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
3.8-22	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
3.8-23	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
3.8-24	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
3.8-25	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
3.8-26	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
3.8-27	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
3.8-28	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
3.8-29	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
3.8-30	This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.8

CHANGE NUMBER

JUSTIFICATION

- 3.8-31 ITS SR 3.8.3.6 would be moved to the ITS Bases. This SR is a preventative-type SR. Sediment in the tank, or failure to perform this SR, does not necessarily result in an inoperable storage tank, as stated in the NUREG-1431, SR 3.8.3.6 Bases. Preventive maintenance SRs, such as the DG inspection SR, generally have been relocated from the CTS.
- ITS SR 3.8.3.3 (DFO testing) and the limits of the Diesel Fuel Oil Testing Program ensure tank sediment is minimized. In addition, ITS SR 3.8.3.1 (fuel oil volume verification) once per 31 days ensures that any gross degradation of the tank wall surface that results in a fuel oil volume reduction is detected and corrected. Further, other government agencies' regulations govern the maintenance of underground storage tanks.
- This change is consistent with industry Traveler TSTF-2.
- 3.8-32 ITS 3.8.1 Required ACTION A:2. and the note preceding Required ACTION D.1. and D.2., would be deleted in accordance with plant system configuration. The 500kV offsite circuit normally supplies power to all three Class 1E buses. The 230kV startup circuit remains available for backup through the automatic load transfer. Therefore, a loss of one offsite power circuit will not result in offsite power to the bus being unavailable.
- 3.8-33 This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.8-34 The frequency of SR 3.8.4.1 and SR 3.8.6.1 would be changed from 7 days to 31 days in accordance with the recommended frequency of at "least monthly" identified in IEEE 450 - 1995. This change is consistent with TSTF-115.
- 3.8-35 The actions of ITS 3.8.2, ITS 3.8.5, ITS 3.8.8, and ITS 3.8.10 would be changed to add a note stating that ITS LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 is not applicable. Irradiated fuel assembly movement also can occur in MODE 1, 2, 3, or 4. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown. This is consistent with industry Traveler TSTF-36.
- 3.8-36 This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.8-37 This change is not applicable to DCPD. See Conversion Comparison Table (Enclosure 6B).
- 3.8-38 ITS SR 3.8.4.7, Note 1, would be revised to allow the modified performance discharge test in ITS SR 3.8.4.8 to be performed in lieu of the service test in ITS SR 3.8.4.7 at any time. This is consistent with IEEE-450 and current licensing basis. IEEE-450 states, "A modified performance test can be used in lieu of a service test at any time." The modified performance test is a test that envelopes the service test with respect to the discharge current being greater than or equal to the service test, and the amp-hours removed is larger. This change is consistent with TSTF-115.
- 3.8-39 ITS SR 3.8.4.6 would be revised for current licensing basis. The restriction to not perform this SR in MODE 1, 2, 3, or 4 would be deleted. The battery charger test is not required to be performed during shutdown since the backup chargers could be in service, maintaining the DC subsystem OPERABLE during performance of this SR.
- 3.8-40 Several SRs would be revised for current licensing basis. ITS SR 3.8.1.2, SR 3.8.1.7, SR 3.8.1.15, and SR 3.8.1.20 would be revised to add a requirement to verify DG achieves proper speed within 10 seconds after start signal.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3/4.8

CHANGE NUMBER

JUSTIFICATION

- 3.8-41 The phrase, "that could degrade battery performance," would be added to clarify that the purpose of the battery inspection is to look for damage to or degradation of the battery that would affect the OPERABILITY of the battery, and that any damage or degradation that does not affect battery operation would not fail the surveillance acceptance criteria. This change is consistent with industry Traveler TSTF-38.
- 3.8-42 This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.8-43 For one DG inoperable while in MODE 1, 2, and 3, the requirement the CTS to confirm the OPERABILITY of the turbine-driven auxiliary feedwater (TDAFW) pump is part of ITS LCO 3.8.1, ACTION B.2. and a note has been added to make this requirement clear.
- 3.8-44 Not Used.
- 3.8-45 This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.8-46 This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
- 3.8-47 The ITS LCO 3.8.3, CONDITION B. requirement for a "per diesel generator" lube oil storage system is revised to reflect the current design of a shared system between units. This current design is similar to the DFO storage system and sizing is based upon a percentage of the DFO usage calculation during the mitigation of a DBA. This calculation reflects meeting a 7 day (or 6 day) operating criteria with single failure and minimum ESF loads.

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(10 Pages)

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.8

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
3.8-01	ITS LCO 3.8.1b. and Conditions E., H., I., and J. have been revised for the DCPD plant-specific three Class 1E vital buses and three DG configuration.	Yes	No	No	No
3.8-02	In accordance with DCPD current licensing bases, the 6 day maximum ACTION statement entry would be revised to 10 days as a limit for the total time of not meeting ITS LCO 3.8.1.	Yes	No	No	No
3.8-03	In accordance with CTS, ITS LCO 3.8.1, Required ACTION B.4. Completion Time to restore DG to OPERABLE would be 7 days.	Yes, 7-day AOT established by License Amendments 44 and 43, issued October 4, 1989.	No, not in CTS.	No, not in CTS.	No, not in CTS.
3.8-04	In accordance with CTS, additional ITS LCO 3.8.1c., Conditions F. and G., would be added to separately address the condition of DFO transfer system inoperabilities.	Yes	No, not in CTS.	No, not in CTS.	No, not in CTS.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.8

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
3.8-05	Deleting NUREG-1431 TS 5.6.7, "EDG Failure Report," and Table 3.8.1-1, "Diesel Generator Test Schedule," and the associated changes to ITS SR 3.8.1.2 and ITS SR 3.8.1.3 is consistent with the recommendations of NRC GL 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," dated May 31, 1994.	Yes	Yes, deleted from CTS by LA 35/21.	Yes, deleted from CTS by LA 101.	Yes
3.8-06	ITS SR 3.8.1.8 and ITS SR 3.8.1.12 would be modified for DCP's system configuration and current licensing basis (CTS 4.8.1.1.1b.1) and CTS 4.8.1.1.1b.2)).	Yes	No	No	No
3.8-07	ITS SR 3.8.1.20 is to be performed once in 10 years and involves simultaneous starting of DGs from standby condition. It is more conservative to perform this surveillance while the unit is shutdown. This is consistent with the CPSES CTS.	No	Yes	No	No
3.8-08	ITS SR 3.8.1.17 would be revised to reflect the design of the DCP DGs for response to an actual or simulated SI signal while operating in test MODE (CTS 4.8.1.1.2b.11)).	Yes	No	No	No
3.8-09	ITS SR 3.8.1.18 would be revised to reflect the design and terminology of the DCP load sequence timers (CTS 4.8.1.1.2b.3).	Yes	No	No	No

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
3.8-10	ITS LCO 3.8.3 would be revised to reflect plant-specific DFO storage and transfer system design and licensing basis.	Yes	Yes	No, not in CTS.	No, not in CTS.
3.8-11	To address the Conditions required by performance of DFO tank cleaning in accordance with DCPD CTS, a note would be added to ITS LCO 3.8.3.	Yes	No	No	No
3.8-12	ITS LCO 3.8.3, Condition E., Required ACTION E.1., and SR 3.8.3.4 for the DG air start subsystem have been revised to reflect plant-specific air start system design and licensing basis. [Each DG has two separate 100% capacity air start systems. Only 1 of the 2 air start receivers is necessary to provide the 3 DG starts required by DCPD licensing basis.]	Yes	Yes	Yes	Yes
3.8-13	ITS specifications for DC distribution, sources and batteries would be revised for DCPD's plant configuration of three Class 1E DC electrical power subsystems consisting of the following major components: 3 60-cell, 125 V batteries; 3 separate 125 Vdc power distribution switchgear assemblies; and 5 battery chargers.	Yes	No	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.8

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
3.8-14	ITS SR 3.8.6.2 is revised from 24 hours to be consistent with current licensing basis, which includes a frequency of once per 7 days. This frequency is adequate to meet the intent of the IEEE 450 recommendations regarding inspections following a severe discharge or overcharge.	Yes	Yes	Yes	Yes
3.8-15	ITS specifications for power sources and power distribution are revised to reflect the DCPD CTS. DCPD's design provides: 3 AC, 3 DC, and 4 AC vital (with 4 inverters) bus electrical power distribution subsystems.	Yes	No	No	No
3.8-16	ITS SR 3.8.4.8 would be revised to be consistent with current licensing basis, which requires an 18-month frequency for testing when the battery shows degradation. This is consistent with the CTS.	Yes	Yes	Yes	Yes
3.8-17	ITS SR 3.8.2.1 would be revised to list only applicable SRs and include as exceptions in the note those SRs that are applicable but should not be performed in MODE 5 or 6.	Yes	Yes	No, maintaining CTS.	Yes
3.8-18	ITS LCO, Conditions, and SRs revised to include DCPD plant-specific requirement for DG turbocharger air assist air receiver pressure requirements.	Yes	No	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.8

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
3.8-19	In accordance with CTS, ITS SR 3.8.1.13 would be modified to include the DG high jacket coolant temperature trip.	No, not CTS.	No, not in CTS.	Yes	Yes
3.8-20	Consistent with industry Traveler TSTF-8, this change would eliminate the potential confusion that may arise, with respect to the application of an unplanned event which satisfies the requirements of a given SR.	Yes	Yes	Yes	Yes
3.8-21	ITS LCO 3.8.7 is revised to delete the detailed description of how the CPSES vital buses are energized.	No	Yes	No	No
3.8-22	ITS SR 3.8.1.9, ITS SR 3.8.1.10, ITS SR 3.8.1.14 DG power factor requirements would be deleted to CPSES reflect current licensing basis.	No	Yes	No	No
3.8-23	ITS SR 3.8.1.9 would be deleted, and ITS SR 3.8.1.10 would be revised to add an electrical frequency requirement in accordance with CTS.	No, not in CTS.	No, not in CTS.	Yes	Yes

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
3.8-24	ITS LCO 3.8.1 Required ACTIONS B.3.2 would be revised to add a note that ITS SR 3.8.1.2 does not need to be performed if the OPERABLE DG has undergone an automatic start and sequence loading or is already operating, in accordance with CTS.	No, not in CTS.	Yes	Yes	Yes
3.8-25	ITS LCO 3.8.1 Conditions would be revised to add a requirement for the blackout sequencer, consistent with CPSES CTS.	No	Yes	No	No
3.8-26	The 2-hour overload part of the 24-hour DG load run would be deleted. ITS SR 3.8.1.14 would be modified to required that if auto-connected loads increase above 2601 kW, 2 hours of the 24-hour run are with the DG loaded to an indicated greater than or equal to 6600 to 6821 kW, consistent with current licensing basis.	No, not in CTS.	No, not in CTS.	Yes	Yes
3.8-27	ITS Table 3.8.6-1 would be revised to change the battery float voltage value and incorporate allowed voltage variation based on the Wolf Creek plant specific design for the AT&T batteries.	No	No	Yes	No
3.8-28	ITS SR 3.8.1.9c. frequency requirement would be deleted, consistent with CPSES CTS.	No	Yes	No	No

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.8

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
3.8-29	ITS LCO 3.8.2, LCO 3.8.5, and LCO 3.8.8, and LCO 3.8.10, Applicability of "during movement of irradiated fuel assemblies," would be deleted consistent with CTS.	No, requirement added in ITS.	Yes	Yes	Yes
3.8-30	ITS SR 3.8.1.4 would be revised consistent with Wolf Creek CTS 4.8.1.1.2a.1 to replace the requirement to verify day tank level with a requirement to verify that the fuel oil transfer pump starts on low standpipe level.	No	No	Yes	No
3.8-31	Consistent with industry Traveler TSTF-2, ITS SR 3.8.3.6 would be moved to the ITS Bases.	Yes	Yes	No, maintaining CTS.	Yes
3.8-32	ITS 3.8.1 Required ACTION A.2. and note preceding Required ACTIONS D.1. and D.2. would be deleted in accordance with DCPD plant system configuration.	Yes	No	No	No
3.8-33	ITS LCO 3.8.3, Condition B., and ITS SR 3.8.3.2 would be revised to note use of a dipstick level for verification of lube oil inventory.	No	Yes	No	No
3.8-34	The frequency of SR 3.8.4.1 and SR 3.8.6.1 would be changed from 7 days to 31 days in accordance with the recommended frequency of at "least monthly" identified in IEEE 450 - 1995.	Yes	Yes	Yes	Yes

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
3.8-35	The ACTIONS of ITS 3.8.2, ITS 3.8.5, ITS 3.8.8, and ITS 3.8.10 would be changed to add a note stating that ITS LCO 3.0.3 is not applicable. This is consistent with industry Traveler TSTF-36.	Yes	No, affected specification are not applicable for fuel movement.	No, affected specification are not applicable for fuel movement.	No, affected specification are not applicable for fuel movement.
3.8-36	A note would be added in accordance with CTS for CPSES that allows one SI sequencer channel to be bypassed for up to 4 hours for surveillance testing provided the other channel is OPERABLE.	No	Yes	No	No
3.8-37	The [ACTUATION LOGIC TEST SR] from CTS 3/4.3 would be added to the ITS.	No, not in CTS.	Yes	Yes	Yes
3.8-38	SR 3.8.4.7, Note 1, would be revised to allow the modified performance discharge test in SR 3.8.4.8 to be performed in lieu of the service test in SR 3.8.4.7 at any time. This change is consistent with TSTF-115.	Yes	Yes	Yes	Yes
3.8-39	ITS SR 3.8.4.6 would be revised for current licensing basis. The restriction to not perform this SR in MODE 1, 2, 3, or 4 would be deleted.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.8

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
3.8-40	ITS SR 3.8.1.7, SR 3.8.1.15, and SR 3.8.1.20 would be revised per DCPD CTS to add a requirement to verify DG achieves proper speed within 10 seconds after start signal.	Yes	No	No	No
3.8-41	The phrase, "that could degrade battery performance," would be added to clarify the purpose of the battery inspection. This change is consistent with industry Traveler TSTF-38.	Yes	Yes	Yes	Yes
3.8-42	An ACTION is added to the LCO for an inoperable automatic load sequencer to immediately declare the affected DG and offsite circuit inoperable.	No, not in CTS.	No, not in CTS.	Yes	Yes
3.8-43	A note is added to ITS 3.8.1, ACTION B.2. to state that the TDAFW is a required redundant feature for one DG inoperable in MODE 1, 2, and 3 consistent with the CTS.	Yes	Yes	Yes	Yes
3.8-44	Not Used.	N/A	N/A	N/A	N/A
3.8-45	The specifications for the power sources and power distribution systems are revised to retain the CTS requirement that one train (subsystem) shall be OPERABLE when shutdown.	No, see CN 3.8-13 and 3.8-15.	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3/4.8

DIFFERENCE FROM NUREG-1431		APPLICABILITY			
NUMBER	DESCRIPTION	Diablo Canyon	Comanche Peak	Wolf Creek	Callaway
3.8-46	Footnote (c) of ITS Table 3.8.6-1 would be modified to retain CTS requirements for using charging current as a substitute for specific gravity measurements.	No	No	Yes	No
3.8-47	The ITS LCO 3.8.3, Condition B., requirement for a "per diesel generator" lube oil storage system is revised to reflect the current design of a shared system between units.	Yes	No	No	No

JLS Conversion to Improved Technical Specifications Diablo Canyon Power Plant

Docket # 50-275
Accession # 4706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 3/4.9 - Refueling Operations

ITS 3.9 - Refueling Operations .



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.9

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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(4 Pages)
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CROSS-REFERENCE TABLE FOR 3/4.9
Sorted by Current TS (CTS)

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.9.1	LCO		01-01-M 01-02-LG 01-04-LG	3.9.1	LCO		
3.9.1	APP		01-05-A	3.9.1	APP		
3.9.1	ACTION		01-06-LS1	3.9.1	ACTION	A	
4.9.1.1	SR		01-09-LS2			Not Used	
4.9.1.2	SR		01-10-LG	3.9.1.1	SR		
3.9.2	LCO		02-01-LS21	3.9.3	LCO		
3.9.2	LCO			3.9.3	ACTION	C	3.9-3
3.9.2	APP			3.9.3	APP		
3.9.2	ACTION	a		3.9.3	ACTION	A	3.9-5
3.9.2	ACTION	b	02-02-M	3.9.3	ACTION	B	3.9-4
4.9.2	SR	a		3.9.3.1	SR		
4.9.2	SR	b, c	02-03-LS3			Not Used	
4.9.2	SR	New	02-03-LS3	3.9.3.2	SR		
3.9.3	LCO		03-01-LG			Not Used	
3.9.3	APP		03-01-LG			Not Used	
3.9.3	ACTION		03-01-LG			Not Used	
4.9.3	SR		03-01-LG			Not Used	
3.9.4	LCO		04-01-LG 04-09-LS14 04-10-LS20	3.9.4	LCO		3.9-7 3.9-11
3.9.4	APP		04-08-LG	3.9.4	APP		
3.9.4	ACTION		04-08-LG	3.9.4	ACTION	A	
4.9.4	SR	a	04-02-LS4 04-10-LS20	3.9.4.1	SR		
4.9.4	SR	New	04-03-LS5	3.9.4.2	SR		
4.9.4	SR	b	04-04-TR1	3.9.4.2	SR		

CROSS-REFERENCE TABLE FOR 3/4.9
Sorted by Current TS (CTS)

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.9.5	LCO		05-01-R			Not Used	
3.9.6	LCO		06-01-R			Not Used	
3.9.7	LCO		07-01-R			Not Used	
3.9.8.1	LCO		08-03-LS6	3.9.5	LCO		3.9-8
3.9.8.1	APP			3.9.5	APP		
3.9.8.1	ACTION		08-01-A	3.9.5	ACTION	A	
4.9.8.1.1	SR			3.9.5.1	SR		3.9-9
4.9.8.1.2	SR			3.9.5.2	SR		3.9-9
3.9.8.2	LCO		08-04-A	3.9.6	LCO		
3.9.8.2	APP			3.9.6	APP		
3.9.8.2	ACTION	a		3.9.6	ACTION	A	
3.9.8.2	ACTION	b		3.9.6	ACTION	B	
4.9.8.2.1	SR			3.9.6.1	SR		3.9-9
4.9.8.2.2	SR			3.9.6.2	SR		3.9-9
	SR	New	08-06-M	3.9.6.3	SR		
3.9.9	LCO		09-01-A	3.9.4	LCO		
3.9.9	APP			3.9.4	APP		
3.9.9	ACTION		09-02-LS7			Not Used	
4.9.9	SR		09-03-LS8			Not Used	
3.9.10.1	LCO			3.9.7	LCO		
3.9.10.1	APP		10-03-LS18	3.9.7	APP		3.9-10
3.9.10.1	ACTION		10-03-LS18	3.9.7	ACTION	A	3.9-2
4.9.10.1	SR		10-02-LS22	3.9.7.1	SR		

CROSS-REFERENCE TABLE FOR 3/4.9
Sorted by Current TS (CTS)

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.9.10.2	LCO					Not Used	
3.9.11	LCO			3.7.15	LCO		
3.9.11	APP		11-01-R	3.7.15	APP		
3.9.11	ACTION		11-02-LS10 11-04-LG	3.7.15	ACTION	A	
4.9.11	SR		11-01-R	3.7.15.1	SR		
3.9.12	LCO			3.7.13	LCO		
3.9.12	APP		12-01-R	3.7.13	APP		
3.9.12	ACTION	a	12-09-LG	3.7.13	ACTION	A	
3.9.12	ACTION	a	12-02-LG	3.7.13	ACTION	C	
3.9.12	ACTION	b	12-02-LG	3.7.13	ACTION	D	
3.9.12	ACTION	c	12-03-A			Not Used	
4.9.12	SR	a		3.7.13.1	SR		
4.9.12	SR	b	12-04-A	3.7.13.2	SR	Filter Program	
4.9.12	SR	c	12-04-A	3.7.13.2	SR	Filter Program	
4.9.12	SR	d.1	12-04-A	3.7.13.2	SR	Filter Program	
4.9.12	SR	d.2	12-05-TR1 12-06-A	3.7.13.3	SR		
4.9.12	SR	d.3	12-08-LS16	3.7.13.4	SR		
4.9.12	SR	e	12-04-A	3.7.13.2	SR	Filter Program	
4.9.12	SR	f	12-04-A	3.7.13.2	SR	Filter Program	
3.9.13	LCO		15-01-R			Not Used	
Fig 3.9-1			15-01-R			Not Used	
3.9.14.1	LCO		14-06-A	3.7.17.2	LCO		

CROSS-REFERENCE TABLE FOR 3/4.9
Sorted by Current TS (CTS)

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.9.14.1	APP			3.7.17.2	APP		
3.9.14.1	ACTION	a	14-03-LS12 14-04-LS13	3.7.17.2	ACTION	A	
3.9.14.1	ACTION	b	14-10-A	3.7.17.2	ACTION	A, NOTE	
4.9.14.1	SR		14-05-LG	3.7.17.2.1	SR		
FIG. 3.9-2				FIG. 3.7.17-2			
3.9.14.2	LCO			3.7.16	LCO		
3.9.14.2	APP		14-07-LS9	3.7.16	APP		
3.9.14.2	ACTION	a	14-08-LS17	3.7.16	ACTION	A	
3.9.14.2	ACTION	b	14-10-A	3.7.16	ACTION	A	
4.9.14.2	SR		14-09-M	3.7.16.1	SR		
3.9.14.3	LCO			3.7.17.1	LCO		
3.9.14.3	APP			3.7.17.1	APP		
3.9.14.3	ACTION	a	14-03-LS12	3.7.17.1	ACTION	A	
3.9.14.3	ACTION	b	14-10-A	3.7.17.1	ACTION	A	
4.9.14.3	SR			3.7.17.1.1	SR		
FIG. 3.9-3				FIG. 3.7.17.1			

CROSS-REFERENCE TABLE FOR 3/4.9
Sorted by Improved TS (ITS)

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.9.1	LCO		01-01-M 01-02-LG 01-04-LG	3.9.1	LCO		
3.9.1	APP		01-05-A	3.9.1	APP		
3.9.1	ACTION		01-06-LS1	3.9.1	ACTION	A	
4.9.1.2	SR		01-10-LG	3.9.1.1	SR		
3.9.2	LCO		02-01-LS21	3.9.3	LCO		
3.9.2	APP			3.9.3	APP		
3.9.2	ACTION	a		3.9.3	ACTION	A	3.9-5
3.9.2	ACTION	b	02-02-M	3.9.3	ACTION	B	3.9-4
3.9.2	LCO			3.9.3	ACTION	C	3.9-3
4.9.2	SR	a		3.9.3.1	SR		
4.9.2	SR	New	02-03-LS3	3.9.3.2	SR		
3.9.4	LCO		04-01-LG 04-09-LS14 04-10-LS20	3.9.4	LCO		3.9-7 3.9-11
3.9.9	LCO		09-01-A	3.9.4	LCO		
3.9.4	APP		04-08-LG	3.9.4	APP		
3.9.9	APP			3.9.4	APP		
3.9.4	ACTION		04-08-LG	3.9.4	ACTION	A	
4.9.4	SR	a	04-02-LS4 04-10-LS20	3.9.4.1	SR		
4.9.4	SR	b	04-04-TR1	3.9.4.2	SR		
4.9.4	SR	New	04-03-LS5	3.9.4.2	SR		
3.9.8.1	LCO		08-03-LS6	3.9.5	LCO		3.9-8
3.9.8.1	APP			3.9.5	APP		
3.9.8.1	ACTION		08-01-A	3.9.5	ACTION	A	
4.9.8.1.1	SR			3.9.5.1	SR		3.9-9
4.9.8.1.2	SR			3.9.5.2	SR		3.9-9

CROSS-REFERENCE TABLE FOR 3/4.9
Sorted by Improved TS (ITS)

<u>Current TS</u>				<u>Improved TS</u>			
<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>	<u>Item</u>	<u>Code</u>	<u>Para.</u>	<u>Change</u>
3.9.8.2	LCO		08-04-A	3.9.6	LCO		
3.9.8.2	APP			3.9.6	APP		
3.9.8.2	ACTION	a		3.9.6	ACTION	A	
3.9.8.2	ACTION	b		3.9.6	ACTION	B	
4.9.8.2.1	SR			3.9.6.1	SR		3.9-9
4.9.8.2.2	SR			3.9.6.2	SR		3.9-9
	SR	New	08-06-M	3.9.6.2	SR		
3.9.10.1	LCO			3.9.7	LCO		
3.9.10.1	APP		10-03-LS18	3.9.7	APP		3.9-10
3.9.10.1	ACTION		10-03-LS18	3.9.7	ACTION	A	3.9-2
4.9.10.1	SR		10-02-LS22	3.9.7.1	SR		

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

Methodology for Cross-Reference Tables
(Continued)

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).

Note: The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
3.9.1	3/4 9-1
3.9.2.	3/4 9-2
3.9.3	3/4 9-3
3.9.4	3/4 9-4
3.9.5	3/4 9-5
3.9.6	3/4 9-6
3.9.7	3/4 9-7
3.9.8.1	3/4 9-8
3.9.8.2	3/4 9-9
3.9.9	3/4 9-10
3.9.10.1	3/4 9-11
3.9.10.2	3/4 9-11a
3.9.11	3/4 9-12
3.9.12	3/4 9-13
3.9.13	3/4 9-15
Figure 3.9-1	3/4 9-16
3.9.14.1	3/4 9-17
Figure 3.9-2	3/4 9-18
3.9.14.2	3/4 9-19
3.9.14.3	3/4 9-20
Figure 3.9-3	3/4 9-22
Methodology	(2 Pages)

* Pages 9-3, 9-5 through 9-7, 9-11a, and 9-15 are intentionally blank

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal and the refueling cavity shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met either: within the limits specified in the COLR

- a. ~~A K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or~~
- b. ~~A boron concentration of greater than or equal to 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.~~

APPLICABILITY: MODE 6*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2,000 ppm, whichever is the more restrictive. action to restore boron concentration to within limits

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

- a. ~~Removing or unbolting the reactor vessel head, and~~
- b. ~~Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.~~

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal and the refueling cavity shall be determined by chemical analysis to be within the limit specified in the COLR at least once each per 72 hours.

~~*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.~~

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE each with continuous visual indication in the control room and one with audible indication in containment and the control room.

02-04-LG

APPLICABILITY: MODE 6.

ACTION:

- a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes except for latching the control rod drive mechanism shaft to the rod cluster control assemblies and friction testing of individual control rods.
- b. With both of the above required monitors inoperable or not operating, ~~determine the boron concentration of the Reactor Coolant System at least once per 12 hours.~~ immediately initiate action to return one source range neutron flux monitor to OPERABLE status, and perform SR (new) every 12 hours.

02-02-M

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

- a. A CHANNEL CHECK at least once per 12 hours.
- b. ~~A CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and~~
- c. ~~A CHANNEL OPERATIONAL TEST at least once per 7 days.~~

02-03-LS3

~~(new) Perform a CHANNEL CALIBRATION every 18 months.*~~

02-03-LS3

~~* Neutron detectors are excluded from CHANNEL CALIBRATION~~

02-03-LS3

3/4.9.3 DECAF TIME

LIMITING CONDITION FOR OPERATION

~~3.9.3 The reactor shall be subcritical for at least 100 hours.~~

~~APPLICABILITY: During movement of irradiated fuel in the reactor vessel.~~

ACTION:

~~With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.~~

SURVEILLANCE REQUIREMENTS

~~4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.~~

REFUELING OPERATIONS

3/4.9.4 CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment penetrations shall be in the following status:

- a. The equipment ~~door~~ hatch closed and held in place by a minimum of four bolts. ED
- b. A minimum of one door in each airlock is closed, and
- c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either: 04-10-LS20
 - 1) Closed by an automatic isolation valve, blind flange, or manual valve, or equivalent, or 04-09-LS14
 - 2) Be capable of being closed by an OPERABLE automatic containment ventilation isolation valve. 04-01-LG

APPLICABILITY: During CORE ALTERATIONS ~~or movement of irradiated fuel within containment, movement of the reactor vessel head over fuel, or movement of the upper internals over fuel.~~ 04-08-LG

ACTION: With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS, movement of irradiated fuel in containment, ~~movement of the reactor vessel head over fuel, or movement of the upper internals over fuel.~~ 04-08-LG

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment penetrations (*) shall be determined to be ~~either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment ventilation isolation valve within 100 hours prior to the start of and at least in the required status once per 7 days during CORE ALTERATIONS, movement of irradiated fuel in containment, movement of the reactor vessel head over fuel, or movement of the upper internals over fuel by:~~ 04-10-LS20

- a- ~~Verifying the penetrations are in their closed/isolated condition, or~~ 04-02-LS4

(new) Verify each required containment purge and exhaust isolation valve actuates to the isolation position by 04-03-LS5

- b- ~~Testing the containment ventilation isolation valves per Specification 4.6.3.2e. at least once per 18 months, by use of an actual or simulated signal.~~ 04-04-TR1

* Except for penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. 04-10-LS20

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

~~3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.~~

~~APPLICABILITY: During CORE ALTERATIONS.~~

ACTION:

~~When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.~~

SURVEILLANCE REQUIREMENTS

~~4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.~~

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

~~3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:~~

- ~~a. The manipulator crane used for movement of fuel assemblies having:
 - 1) A minimum capacity of 3250 pounds, and
 - 2) An overload cut off limit less than or equal to 2700 pounds.~~
- ~~b. The auxiliary hoist used for movement of control rods having:
 - 1) A minimum capacity of 700 pounds, and
 - 2) A load indicator which shall be monitored to preclude lifting loads in excess of 600 pounds.~~

~~APPLICABILITY: During movement of control rods or fuel assemblies within the reactor vessel.~~

~~ACTION:~~

~~With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor vessel.~~

SURVEILLANCE REQUIREMENTS

~~4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to removal of the reactor vessel head by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2700 pounds.~~

~~4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to removal of the reactor vessel head by performing a load test of at least 700 pounds.~~

3/4.9.7 CRANE TRAVEL — FUEL HANDLING BUILDING

LIMITING CONDITION FOR OPERATION

~~3.9.7 Loads in excess of 2500 pounds* shall be prohibited from travel over fuel assemblies in the spent fuel pool.~~

~~APPLICABILITY: With fuel assemblies in the spent fuel pool.~~

ACTION:

- a. ~~With the requirements of the above specification not satisfied, place the crane load in a safe condition.~~
- b. ~~The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

~~4.9.7 Loads shall be verified to be less than 2500 pounds prior to movement over fuel assemblies in the spent fuel pool.~~

~~*The movable fuel handling building wall may travel over fuel assemblies in the spent fuel pool.~~

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) train shall be OPERABLE and in operation.*, **

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is at least 23 feet.

ACTION:

With no RHR train OPERABLE and in operation, immediately suspend all operations involving an increase in the reactor decay heat load loading irradiated fuel assemblies into the core or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR train to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. 08-01-A

SURVEILLANCE REQUIREMENTS

4.9.8.1.1 With the reactor subcritical less than 57 hours, at least one RHR train shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

4.9.8.1.2 With the reactor subcritical for 57 hours or more, at least one RHR train shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 1300 gpm at least once per 12 hours.

* The RHR train may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs, provided no operations are permitted that would cause a reduction of the RCS boron concentration. 08-03-LS6

** The RHR train may be removed from operation and OPERABLE status for up to 2 hours per 8 hour period for the performance of leak testing the RHR suction isolation valves provided no operations are permitted that would cause a reduction of the RCS boron concentration. 08-03-LS6

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) trains shall be OPERABLE and at least one RHR train shall be in operation.*

08-04-A

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

- a. With less than the required RHR trains OPERABLE, immediately initiate corrective action to return the required RHR trains to OPERABLE status, or to establish at least 23 feet of water above the reactor vessel flange, as soon as possible.
- b. With no RHR train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR train to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2.1 With the reactor subcritical less than 57 hours, at least one RHR train shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

4.9.8.2.2 With the reactor subcritical for 57 hours or more, at least one RHR train shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 1300 gpm at least once per 12 hours.

~~(NEW) Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation at least once per 7 days.~~

08-06-M

~~*Prior to initial criticality, the RHR train may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.~~

08-04-A

REFUELING OPERATIONS

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation Isolation System shall be OPERABLE capable of automatic closure or manually isolated.

09-01-A

09-02-LS7

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within containment.

ACTION:

a- ~~With the Containment Ventilation Isolation System inoperable, close each of the ventilation penetrations providing direct access from the containment atmosphere to the outside atmosphere.~~

09-02-LS7

b- The provisions of Specification 3.0.3 are not applicable.

09-01-A

SURVEILLANCE REQUIREMENTS

~~4.9.9 The Containment Ventilation Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment ventilation isolation occurs on a High Radiation test signal from the containment ventilation exhaust radiation monitoring instrumentation channels.~~

09-03-LS8

REFUELING OPERATIONS

3/4.9.10 WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment when the reactor pressure vessel pressure contains irradiated fuel assemblies.

10-03-LS18

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of irradiated fuel assemblies within the reactor pressure vessel containment. The provisions of Specification 3.0.3 are not applicable.

10-03-LS18

SURVEILLANCE REQUIREMENTS

4.9.10.1 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of irradiated fuel assemblies within containment.

10-02-LS22

REFUELING OPERATIONS

WATER LEVEL - REACTOR VESSEL

CONTROL RODS

10-01-R

LIMITING CONDITION FOR OPERATION

~~3.9.10.2 At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor pressure vessel.~~

~~APPLICABILITY: During movement of control rods within the reactor pressure vessel while in MODE 6.~~

ACTION:

~~With the requirements of the above specification not satisfied, suspend all operations involving movement of control rods within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

~~4.9.10.2 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of control rods within the reactor pressure vessel.~~

REFUELING OPERATIONS

3/4.9.11 WATER LEVEL - SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.9.11 ~~At least~~ ~~The spent fuel storage pool water level shall be~~ \geq 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: ~~Whenever~~ ~~During movement of~~ irradiated fuel assemblies are in the spent fuel storage pool.

11-01-LG

ACTION:

- a. With the requirements of the above specification not satisfied, ~~immediately~~ suspend all movement of irradiated fuel assemblies and crane operations with loads in the spent fuel storage pool areas and restore the water level to within its limit within 4 hours.
- b. The provisions of Specification 3.0.3 are not applicable.

11-04-LG

11-02-LS10

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are ~~being moved~~ in the spent fuel pool.

11-01-LG

REFUELING OPERATIONS

3/4.9.12 FUEL HANDLING BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two Fuel Handling Building Ventilation Systems trains shall be OPERABLE.

APPLICABILITY: ~~Whenever~~ ~~During movement of irradiated fuel is assemblies in the spent Fuel pool Handling Building.~~ 12-01-LS24

- ACTION:
- a. ~~With one Fuel Handling Building Ventilation System train inoperable, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the OPERABLE Fuel Handling Building Ventilation System train is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal absorbers.~~ 12-02-LG
12-09-LG
12-04-A
 - b. ~~With no Fuel Handling Building Ventilation System trains OPERABLE, suspend all operations involving movement of irradiated fuel within the spent fuel pool or crane operation with loads over the spent fuel pool until at least one Fuel Handling Building Ventilation System train is restored to OPERABLE status.~~ 12-02-LG
ED
 - c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

~~4.9.12 The above required Fuel Handling Building Ventilation Systems shall be demonstrated OPERABLE:~~

- a. ~~At least once per 31 days by initiating flow through each train of FHBVS the HEPA filters and charcoal absorbers and verifying that the system operates for at least 15 minutes;~~ ED
12-04-A
- b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal absorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:
 - 1) Visually verifying that, with the system operating at a flow rate of 35,750 cfm ± 10% and exhausting through the HEPA filters and charcoal absorbers, the damper valve M-29 is closed;
 - 2) Verifying that the cleanup system satisfies the in place penetration and bypass Leakage testing acceptance criteria of less than 1% and uses the test procedures guidance in ANSI N 510 - 1980, and the system flow rate is 35,750 cfm ± 10%; 12-04-A

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

- 3) Verifying, ~~within 31 days after removal~~, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803 - 1989 at 95% R.H. for a methyl iodide penetration of less than 4.3%; and 12-10-LS9
- 4) ~~Verifying a system flow rate of 35,750 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.~~ 12-11-A
12-10-LS9
- c. After every 720 hours of charcoal absorber operation by verifying ~~within 31 days after removal~~, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803 - 1989 at 95% R.H. for a methyl iodide penetration of less than 4.3%:
- d. At least once per 18 months by: 12-06-A
- 1) Verifying that the pressure drop across the combined HEPA filters and charcoal absorber banks is less than 4.1 inches Water Gauge while operating the system at a flow rate of 35,750 cfm ± 10%.
- 2) ~~Verifying that on a high radiation test~~ actual or simulated actuation signal, ~~the system automatically starts (unless already operating) each item of the train activates and directs its exhaust flow through the HEPA filters and charcoal absorber banks, and~~ 12-05-TR1
12-04-A
- 3) ~~At least once every 18 months on a STAGGERED TEST BASIS~~ Verifying that the system a train maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during system operation. 12-08-LS16
- e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 35,750 cfm ± 10%; and 12-04-A
- f. After each complete or partial replacement of a charcoal absorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon test gas while operating the system at a flow rate of 35,750 cfm ± 10%.

REFUELING OPERATIONS

15-01-R

3/4.9.13 SPENT FUEL SHIPPING CASK MOVEMENT

LIMITING CONDITION FOR OPERATION

~~3.9.13 No spent fuel shipping cask handling operation near the spent fuel pool (i.e., any movement of a cask located north of column line 12.9 for Unit 1 or south of column line 23.1 for Unit 2) shall be performed with fuel in the spent fuel cask exclusion zone identified in Figure 3.9.1.~~

~~APPLICABILITY: During all cask handling operations.~~

ACTION:

~~With the requirements of the above specification not satisfied, move the cask out of the specified area(s), or move spent fuel from all locations in the racks within the exclusion zone identified in Figure 3.9.1.~~

SURVEILLANCE REQUIREMENTS

~~4.9.13 Racks within the exclusion zone identified in Figure 3.9.1 shall be verified to contain no fuel prior to the movement of the cask into the specified area.~~

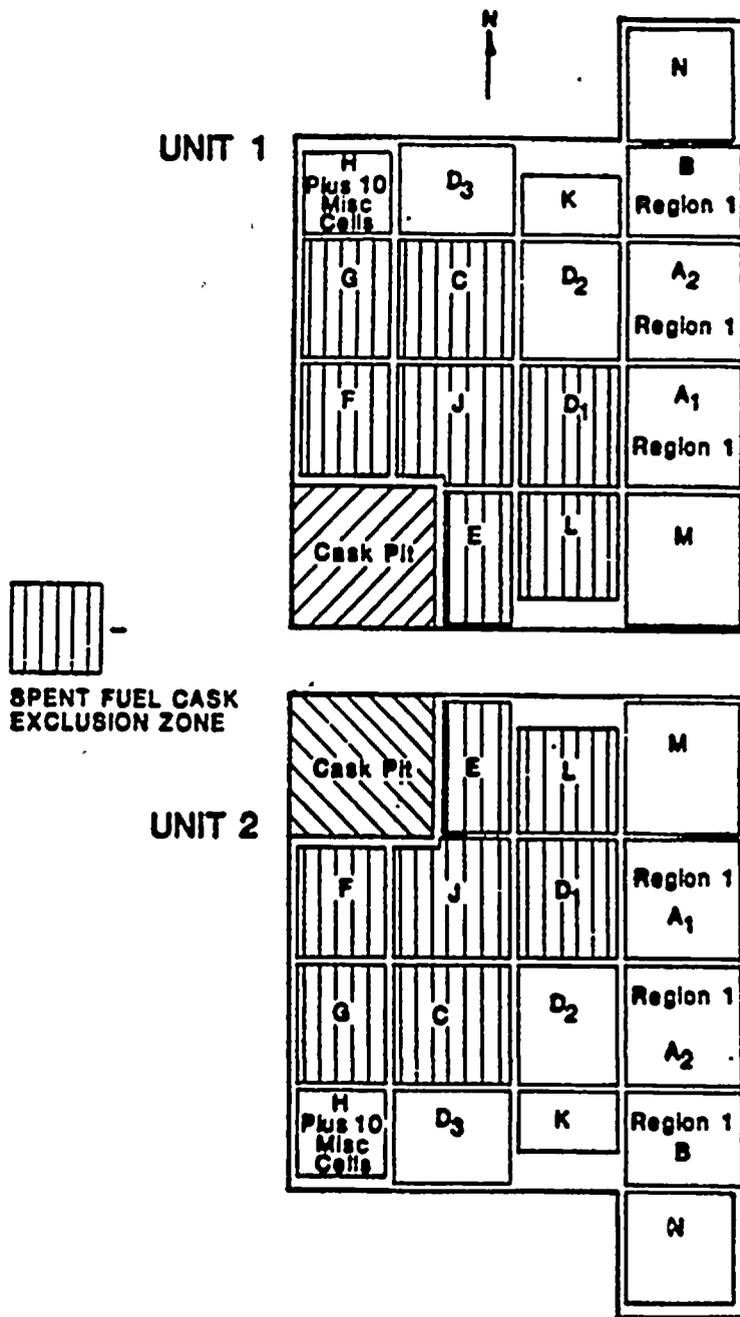


FIGURE 3.9-1 UNITS 1 AND 2 SPENT FUEL POOL LAYOUT

REFUELING OPERATIONS

3/4.9.14 SPENT FUEL ASSEMBLY STORAGE

SPENT FUEL POOL REGION 2

LIMITING CONDITION FOR OPERATION

3.9.14.1 The following conditions shall be met for storage of fuel assemblies in region 2 of the spent fuel pool:

- a. The combination of initial enrichment, fuel pellet diameter, and cumulative burnup of the assemblies is within the acceptable area of Figure 3.9-2; or
- b. The assemblies are put into a checkerboard pattern with water cells or non-fissile material.

APPLICABILITY: Whenever ~~any~~ fuel assemblies are ~~assembly is stored~~ in the spent fuel pool.

ACTION:

- a. With the requirements of the above specification not satisfied, ~~suspend all movement of fuel assemblies and crane operations (with loads in the fuel storage area) except to perform the following: move the non-complying fuel assemblies into compliance with the above Specification or Specification 3.9.14.3 into a pattern that complies with requirements of the above specification. Until the requirements of the above specification and Specification 3.9.14.3 are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.~~ 14-03-LS12
- b. The provisions of Specifications 3.0.3 and 3.0.4 are ~~is~~ not applicable. 14-04-LS13
14-10-A

SURVEILLANCE REQUIREMENTS

4.9.14.1 The cumulative burnup of each spent fuel assembly stored in Region 2 shall be determined by analysis of its burnup history, prior to storage in Region 2. ~~A complete record of initial enrichment, fuel pellet diameter, and the cumulative burnup analysis shall be maintained for the time period that each fuel assembly remains in Region 2 of the spent fuel pool.~~ 14-05-LG

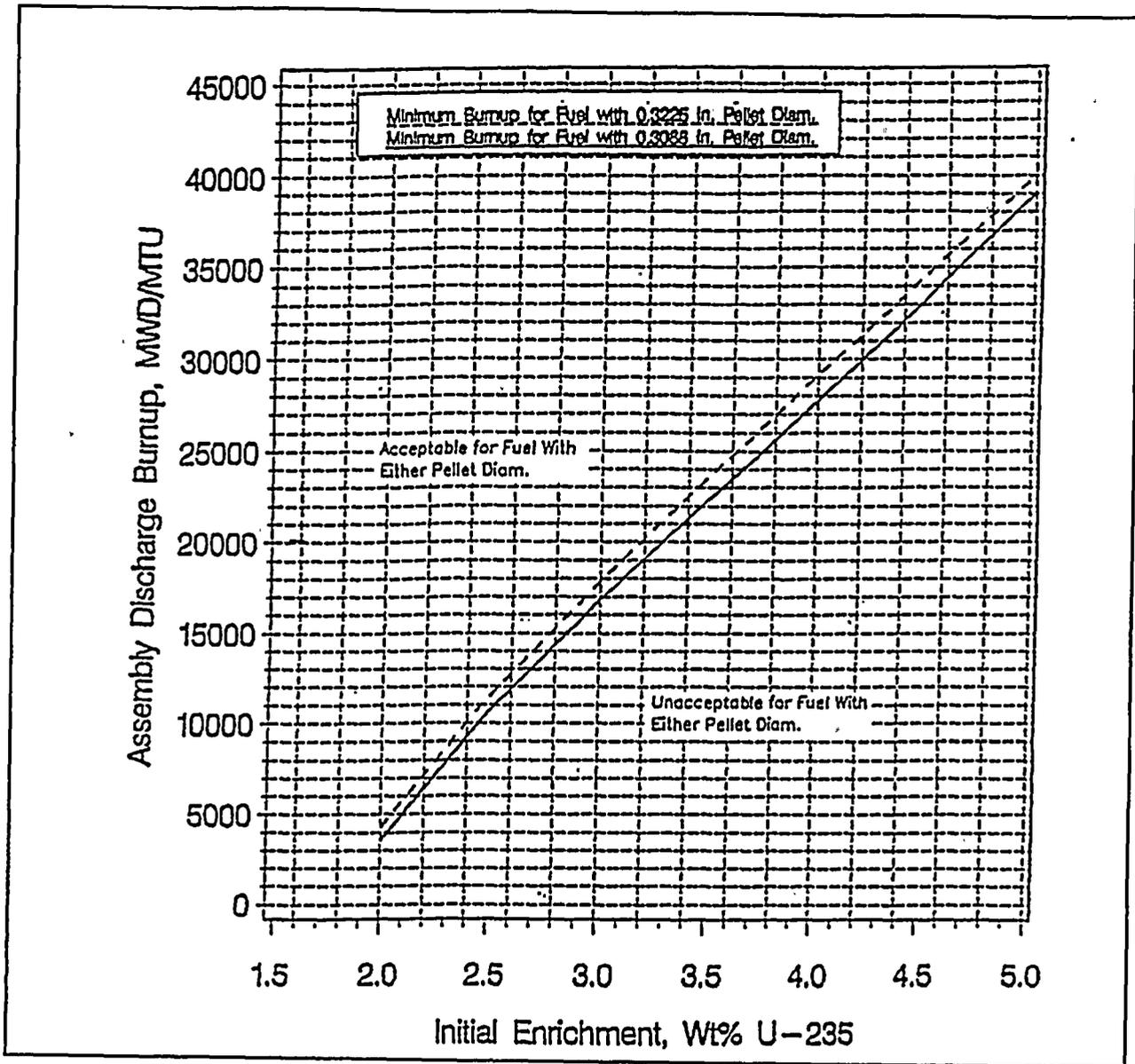


FIGURE 3.9-2
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT
STORAGE IN REGION 2

REFUELING OPERATIONS

SPENT FUEL ASSEMBLY STORAGE

SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.14.2 The boron concentration of the spent fuel pool shall be greater than or equal to 2000 ppm.

APPLICABILITY: Whenever fuel assemblies are in the spent fuel pool .

ACTION:

- a. With the requirements of the above specification not satisfied, immediately suspend all movement of fuel assemblies in the spent fuel pool -and initiate corrective actions to restore the boron concentration.
- b. The provisions of Specifications ~~3.0.3 and 3.0.4~~ are not applicable.

14-10-A

SURVEILLANCE REQUIREMENTS

4.9.14.2 The ~~Verify the~~ boron concentration of the spent fuel ~~storage~~ pool shall ~~be determined by chemical analysis~~ at least once per 31 days.

01-10-LG

REFUELING OPERATIONS

SPENT FUEL ASSEMBLY STORAGE

SPENT FUEL POOL REGION 1

LIMITING CONDITION FOR OPERATION

3.9.14.3 The following conditions shall be met for storage of fuel assemblies in Region 1 of the spent fuel pool:

- a. The initial enrichment is 4.5 weight percent U-235 or less; or
- b. The initial enrichment is from 4.5 up to a maximum of 5.0 weight percent U-235, and any of the following conditions are met:
 - 1) The combination of initial enrichment and cumulative burnup of the assemblies is within the acceptable area of Figure 3.9-3; or
 - 2) The assemblies initially contained a minimum of a nominal 36 mg/in. per assembly of the isotope B-10 integrated in the fuel rods; or
 - 3) The assemblies are put in a checkerboard pattern with any of the following:
 - a) water cells, or
 - b) assemblies that initially contained a minimum of a nominal 72 mg/in. per assembly of the isotope B-10 integrated in the fuel rods, or
 - c) partially irradiated fuel of at least 8000 MWD/MTU cumulative burnup; or
 - 4) The assemblies are put into a pattern with alternate rows of fuel assemblies and water cells.

APPLICABILITY: Whenever fuel assemblies are in Region 1 of the spent fuel pool.

ACTION:

- a. With the requirements of the above specification not satisfied, 14-03-LS12
~~suspend all movement of fuel assemblies and crane operations (with loads in the fuel storage area) except to perform the following: move the non-complying fuel assemblies into a pattern that complies with requirements of the above specification or Specification 3.9.14.1. Until the requirements of the above specification and Specification 3.9.14.1 are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.~~
- b. The provisions of Specifications 3.0.3 and 3.0.4 are ~~is~~ not applicable. 14-10-A

REFUELING OPERATIONS

SPENT FUEL ASSEMBLY STORAGE

SPENT FUEL POOL REGION 1

SURVEILLANCE REQUIREMENTS

4.9.14.3 The cumulative burnup of each fuel assembly stored in Region 1 shall be determined by analysis of its burnup history, prior to storage in Region 1.—~~A complete record of initial enrichment, initial integral boron content, and the cumulative burnup analysis shall be maintained for the time period that the fuel assembly remains in Region 1 of the spent fuel pool.~~

14-05-LG

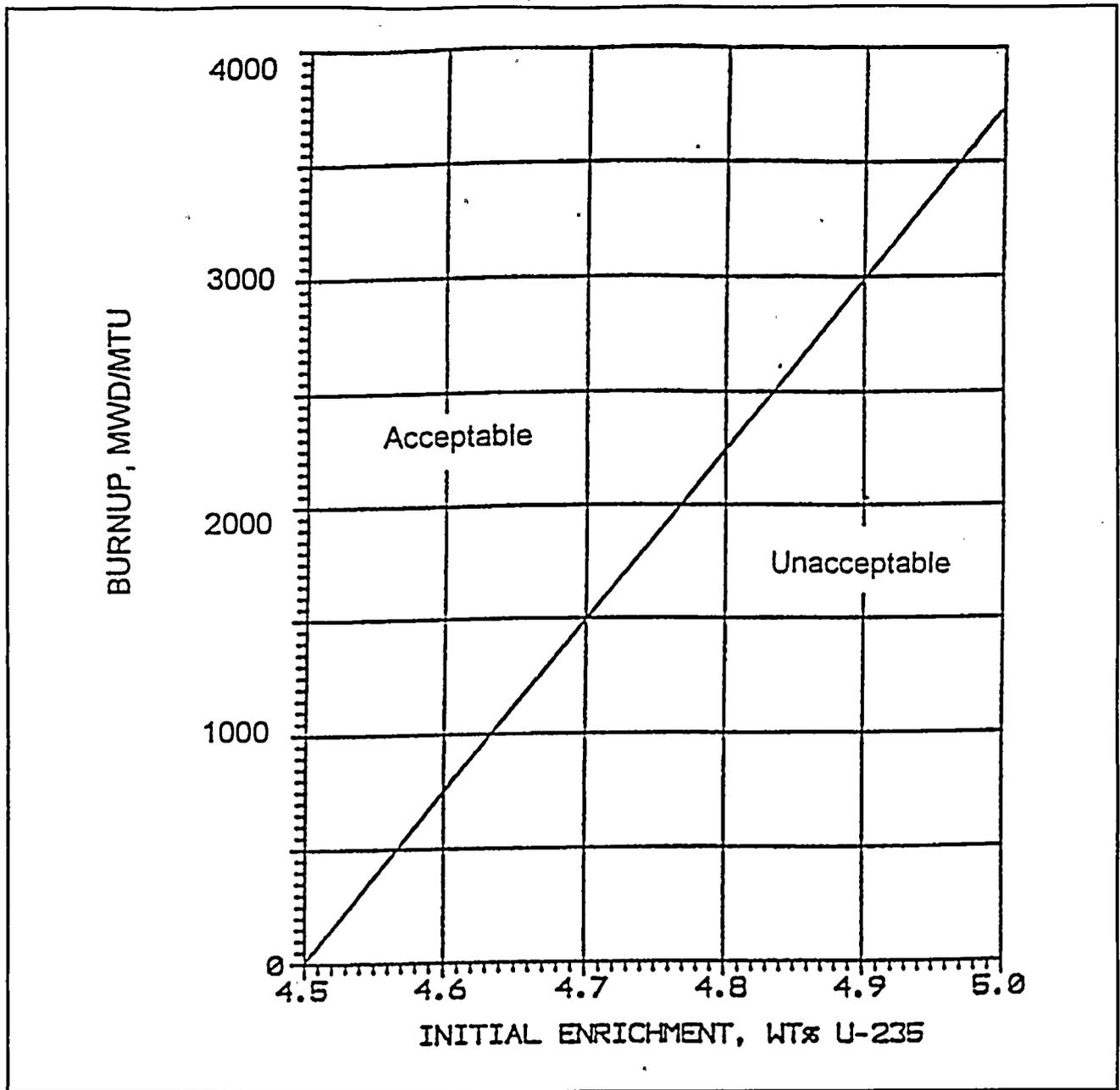


FIGURE 3.9-3
 MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
 AS A FUNCTION OF INITIAL ENRICHMENT (NO IFBA) TO PERMIT
 STORAGE IN REGION 1(Tech Spec pages HERE)(Tech Spec pages HERE)

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. Deletions - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. Additions - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. Modifications - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. Administrative - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

Methodology For Mark-Up of Current TS
(Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers (1 Page)

Description of Changes (7 Pages)

TECHNICAL CONVERSION CHANGE NUMBERS

SECTION 3/4.9

Technical Specification Title	CHG. NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Boron Concentration	01	3.9.1	3.9.1	3.9.1	3.9.1
Instrumentation	02	3.9.2	3.9.2	3.9.2	3.9.2
Decay Time	03	3.9.3	3.9.3	3.9.3	3.9.3
Containment Building Penetrations	04	3.9.4	3.9.4	3.9.4	3.9.4
Communications	05	None	None	3.9.5	3.9.5
Manipulator Crane	06	None	None	3.9.6	3.9.6
Crane Travel - Spent Fuel Stor. Building	07	None	None	3.9.7	3.9.7
RHR and Coolant Recirculation - High Water Level	08	3.9.8.1	3.9.8.1	3.9.8.1	3.9.8.1
RHR and Coolant Recirculation - Low Water Level	08	3.9.8.2	3.9.8.2	3.9.8.2	3.9.8.2
Containment Purge and Exhaust Isolation System	09	3.9.9	3.9.9	None	3.9.9
Water Level Reactor Vessel	10	3.9.10.1	3.9.10.1	3.9.9.1	3.9.10.1
Water Level -Storage Pool	11	3.9.11	3.9.11	3.9.10	3.9.11
Fuel Storage Pool Air Cleanup System	12	3.9.13	3.9.13	None	3.9.12
Water Level Reactor Vessel - Control Rods	10	None	None	3.9.9.2	3.9.10.2
Spent Fuel Assembly Storage	14	3.9.12	3.9.12	None	3.9.14
Spent Fuel Shipping Cask Movement	15	None	None	None	3.9.13 LAR submitted to relocate

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	A	Adds the filled portion of the refueling cavity to the locations in which the boron concentration must be maintained and specifies that the concentration must be maintained in locations connected to the RCS.
01-02	LG	Specifies that the required limits for the boron concentration will be moved to the Core Operating Limits Report, in accordance with NUREG-1431. This change removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.
01-03	LG	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-04	LG	The specified limit on $k_{eff} \leq 0.95$ is moved to the Bases; however, the limit is effectively maintained by the requirement to keep boron concentration within limits which remains in the LCO. As noted in 1-02-LG above, the boron concentration limit will be maintained in the COLR. This change removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.
01-05	A	The footnote defining the "REFUELING" condition is not necessary because it duplicates the definition of MODE 6 in ITS Table 1.1-1. This change does not result in a change to technical requirements and is consistent with NUREG-1431.
01-06	LS1	The requirements to initiate boration at a specified flow rate having a specified boron concentration is replaced by the more general requirement to initiate boration to restore the required boron concentration. The reactor operators are expected to select the best method of increasing the boron concentration to the required value specified in the COLR. The proposed change clarifies that action is applicable only to restoring boron concentration to within limit. This change is acceptable because it is an example of removing procedural details while maintaining the actual limiting condition as a TS requirement.
01-07	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-08	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-09	LS2	The SR to verify reactivity conditions is deleted, as it is generally descriptive of the MODE 6 conditions, as defined in NUREG-1431, and is addressed by SR 4.9.1.2. This change is acceptable because the boron concentration is required to be within limit prior to entry into MODE 6 in accordance with the Applicability Note for ITS 3.9.1. Thus, the deleted SR is redundant to other requirements that remain in TS.
01-10	LG	Moves the description in the SR to determine the boron concentration by chemical analysis to the Bases. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.
01-11	LS19	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
01-12	LG	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-01	LS21	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-02	M	The ACTION statement is revised to require that restoration of one monitor is immediately initiated. This change adds a more stringent TS requirement which is appropriate and consistent with NUREG-1431.
02-03	LS3	The ANALOG CHANNEL OPERATIONAL TEST (ACOT) requirements are deleted and a Channel Calibration is added, in accordance with NUREG-1431. In Mode 6, the source range monitors are required for indication only and there are no precise set points associated with these instruments. In this capacity, the source range instrumentation is typically used to read a relative change in count rate. The source range instrumentation is monitored for significant changes in count rate which are important to evaluate the change in core status. Even the accepted convention defining criticality only requires a slowly increasing count rate be verified. Consistent with NUREG-1431, indicating instruments only require channel checks and channel calibrations. The more frequent ACOTs are applied only to those channels with operational interlocks or other set point actuations. Therefore, the MODE 6 channel checks and channel calibration requirements for the source range monitors are adequate to assure their operability, considering the more frequent ACOTs performed on this instrumentation in other Modes, the effectiveness of these surveillance requirements in maintaining other indicating instruments operable, and the accuracy required of these instruments in MODE 6.
02-04	LG	Consistent with NUREG-1431, the requirements related to indication provided by the source range detectors would be moved to the Bases.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
3-01	R	Consistent with NUREG-1431, the subcriticality requirement prior to irradiated fuel movement is relocated to a licensee controlled document. This change is acceptable based on the schedule requirements following shutdown to attain plant conditions for movement of irradiated fuel. These schedule requirements provide assurance that the requirements of the decay time LCO would not be exceeded.
04-01	LG	This change removes the word "automatic" from the requirement that each penetration be capable of being closed by an OPERABLE automatic containment purge isolation valve. The requirement for an automatic valve would be stated in the Bases. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.
04-02	LS4	Removes Surveillance requirement to perform verification within 100 hours prior to the start of core alteration or movement of irradiated fuel. This is consistent with NUREG-1431, and is acceptable because the deleted requirement is redundant with the requirement to meet the LCO at the time that CORE ALTERATIONS or fuel movement begins.
04-03	LS5	The frequency of verifying that the [Containment Purge and Exhaust isolation] occurs is changed from 7 days to 18 months. This is consistent with NUREG-1431. This change is acceptable because the revised frequency requirement will continue to assure the OPERABILITY of the valves. The new frequency is consistent with those SRs applicable to ESFAS-type functions and in service valve testing which are appropriate for the containment isolation function.
04-04	TR1	Revised Surveillance requirement to allow for increased flexibility in using an actual or simulated actuation signal. Identification of specific signals is moved to the Bases.
04-05	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-06	LS23	Not applicable to DCPP: See Conversion Comparison Table (Enclosure 3B).
04-07	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-08	LG	Relocate the references to Heavy Loads in the Applicability and ACTION section of LCO 3.9.4 to the FSAR.
04-09	LS14	LCO 3.9.4 would be modified to permit an approved functional equivalent of a valve or blind flange to isolate containment penetrations. This is consistent with NUREG-1431.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
04-10	LS20	Adds a footnote stating that penetration flow paths that provide direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. This change is consistent with traveler WOG-76 and with previously approved administrative controls for personnel air locks.
05-01	R	This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
06-01	R	This change relocates the CTS section for the Manipulator Crane to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
07-01	R	This change relocates the current section dealing with crane travel to Licensee controlled documents as part of the conversion of the CTS to the format and expanded Bases of the ITS.
08-01	A	This change, consistent with NUREG-1431, provides technical guidance and clarification that loading irradiated fuel assemblies in the core is the specific activity of concern that could increase the reactor decay heat load. This is not a technical change because, under these conditions, the only activity that could increase reactor decay heat load is loading irradiated fuel into the reactor vessel.
08-02	A	Not applicable to DCPD. See Conversion Comparison Table (Enclosure 3B).
08-03	LS6	This change allows the removal of the RHR loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs. This allows increased flexibility for core mapping and isolation valve testing. No operations are permitted that would cause a reduction of the RCS boron concentration. This change is consistent with NUREG-1431.
08-04	A	This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred.
08-05	-	Not Used
08-06	M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days. This change adds a more stringent TS requirement which is appropriate and consistent with NUREG-1431.
09-01	A	Most of the requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4. This change does not result in a change to technical requirements and is consistent with NUREG-1431.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
09-02	LS7	Deletes the requirement to close each purge valve when the Containment Ventilation System is inoperable. The ITS only requires that core alterations and irradiated fuel movement be suspended.
09-03	LS8	The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the SR frequency from once per 7 days and within 100 hours prior to CORE ALTERATIONS to once per 18 months. This change is acceptable because it would apply the same 18-month frequency to the containment ventilation valves as applied to other containment isolation valves that must be OPERABLE for accidents more severe than an FHA.
09-04	LS15	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-01	R	This change relocates the CTS requirements concerning reactor vessel water level for movement of control rods.
10-02	LS22	This change deletes the surveillance requirement to verify water level within 2 hours prior to the start of movement of [irradiated] fuel assemblies. This is acceptable because the LCO must be met at the time that movement of [irradiated] fuel assemblies is performed.
10-03	LS18	Revises Applicability such that it applies only when irradiated fuel is moved. The CTS also applies to movement of un-irradiated fuel when there is irradiated fuel in the core.
11-01	LG	This change modifies the Applicability to "during" movement of irradiated fuel assemblies in the fuel storage pool to be consistent with the Fuel Handling Accident (FHA). The portions of this requirement applicable to whenever irradiated fuel is in the fuel storage pool will be moved to a Licensee controlled document. This change is consistent with NUREG-1431, and removes details that are not required to be in the TS to protect the health and safety of the public while retaining the basic limiting conditions for operation.
11-02	LS10	This change modifies the ACTION statement by eliminating the 4 hour time requirement to restore water level. This change, which is consistent with NUREG-1431, is acceptable because the ITS Required Action would suspend movement of irradiated fuel immediately which would establish conditions outside the Applicability of the LCO.
11-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
11-04	LG	This change moves the restriction on crane operation to a licensee controlled document. The restriction on crane operations may be removed because it is not in the assumptions used for the FHA. Crane operations that could adversely affect fuel stored in the spent fuel pool are controlled as analyzed in the review of heavy load movements. This change is consistent with NUREG-1431, and moves requirements that do not meet the criteria for inclusion in the TS.
12-01	LS24	The Applicability would be changed to "During movement of irradiated fuel in the fuel building" instead of "Whenever irradiated fuel is in the spent fuel pool" consistent with NUREG-1431. The proposed Applicability is consistent with the assumptions used in the FHA in the Fuel Handling Building which postulates the inadvertent drop of an irradiated fuel assembly. Potential damage to fuel assemblies due to dropping of heavy loads is addressed by CN 12-02-LG.
12-02	LG	Moves the restriction on crane operations over the spent fuel storage areas when the fuel building air cleanup system was inoperable. The restriction on crane operations may be removed because it is not consistent with the assumptions used for the FHA. Crane operations that could adversely affect fuel stored in the spent fuel pool is prohibited in accordance with plant procedures as analyzed in the review of heavy load movements.
12-03	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
12-04	A	The SR regarding filter testing would be moved to a "Ventilation Filter Testing Program" that is called out in the Administrative Controls Section 5.5.11 of the ITS. This change does not result in a change to technical requirements.
12-05	TR1	Revised SR to allow for increased flexibility in using an actual or simulated actuation signal. Identification of the specific signal is moved to the Bases.
12-06	A	This requirement would have the operability of each train of the [Fuel Handling Building Ventilation System (FHBVS)] (including maintaining negative pressure in the building) to be demonstrated. This is consistent with current practice. This change does not result in a change to technical requirements and is consistent with NUREG-1431.
12-07	LS25	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
12-08	LS16	The proposed change would allow the 18-month testing of the [FHBVS] ability to maintain the required pressure differential between the building and the outside atmosphere to be performed on a STAGGERED TEST BASIS.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
12-09	LG	The requirement for an OPERABLE emergency power source for an OPERABLE FHBVS train is moved to the Bases. This is consistent with NUREG-1431.
12-10	LS9	The "within 31 days after removal" requirement for completion of laboratory analyses is deleted. This requirement is not contained in the ITS nor is it contained in the regulatory guide or ANSI standards.
12-11	A	The SR to measure [FHBVS] flow rate is not identified as a separate SR in the ITS because it is verified during the other in-place filter tests (see ITS 5.5.11 a. and b.). This change does not result in a change to technical requirements.
12-12	LS26	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
14-01	LS11	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
14-02	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
14-03	LS12	This change would delete the ACTION requirements to suspend all other movement of fuel assemblies and crane operations. This change is consistent with NUREG-1431.
14-04	LS13	Deletes the action statement requirement to verify spent fuel pool boron concentration every 8 hours while ACTION is being taken to relocate noncomplying spent fuel assemblies from Region 2 to Region 1. This change is consistent with NUREG-1431.
14-05	LG	The requirement to keep records of the burnup analysis for all assemblies in Region 2 would be relocated to a licensee controlled document. This change is consistent with NUREG-1431, and moves requirements that do not meet the criteria for inclusion in the TS.
14-06		Not used.
14-07		Not used.
14-08		Not used.
14-09		Not used.
14-10	A	The Statement that 3.0.4 is not applicable is deleted. This is consistent with NUREG-1431. This change does not result in a change to technical requirements.
15-01	R	The requirement to empty the spent fuel exclusion zone area prior to any spent fuel shipping cask handling operations is relocated to a Licensee controlled document.

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(8 pages)

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	Adds the filled portion of the refueling cavity to the locations in which the boron concentration must be maintained.	Yes	Yes	Yes	Yes
01-02 LG	Specifies that the required limits for the boron concentration will be in the Core Operating Limits Report (COLR). In addition, the provision to maintain a uniform concentration is discussed in the ITS Bases.	Yes	Yes	Yes	Yes
01-03 LG	Instead of providing the tag numbers of the valves used to isolate unborated water sources, the function of the valves is used. The valve tag numbers are moved to the Bases.	No, Current Technical Specifications (CTS) based upon licensed dilution accident.	Yes	Yes	No, Maintaining valve numbers in CTS
01-04 LG	The specified limit on $k_{eff} \leq 0.95$ is moved to the Bases.	Yes	Yes	Yes	Yes
01-05 A	The footnote defining the "REFUELING" condition is not necessary because it duplicates the definition of MODE 6.	Yes	No, not in CTS	Yes	Yes
01-06 LS1	The requirements to initiate boration at a specified flow rate having a specified boron concentration is replaced by the more general requirement to initiate boration to restore the required boron concentration. Additionally, the ACTION statement is revised to clarify that action is applicable only to boron concentration.	Yes	Yes	Yes	Yes
01-07 M	A new ACTION statement is incorporated that specifies the appropriate activities if the isolation valves for unborated water sources are not secured in the closed position.	No, CTS based upon Licensed Dilution Accident.	No	Yes	Yes
01-08 M	Separate entry into the ACTION is allowed for each unborated water source isolation valve.	No, CTS based upon Licensed Dilution Accident.	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.9

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-09 LS2	The SR to verify reactivity conditions is deleted.	Yes	Yes	Yes	Yes
01-10 LG	Moves the description in the SR to determine the boron concentration by chemical analysis to the Bases.	Yes	Yes	Yes	Yes
01-11 LS19	The time required to verify that the boron concentration is within its limits has been relaxed from 1 hour to 4 hours.	No, CTS based upon Licensed Dilution Accident.	Yes	No, not in CTS	No, not in CTS
01-12 LG	Generalizes the requirement to verify the dilution isolation valves are closed by mechanical stops or removal of motive power.	No, CTS based upon Licensed Dilution Accident.	Yes	No, not in CTS	No, not in CTS
02-01 LS21	The requirements related to indication provided by the source range detectors would be deleted from the LCO.	No, See 02-04-LG	Yes	Yes	Yes
02-02 M	The ACTION statement is revised to require that restoration of one monitor is immediately initiated.	Yes	Yes	Yes	Yes
02-03 LS3	The ANALOG CHANNEL OPERATIONAL TEST requirements are deleted and a channel calibration is added.	Yes	Yes	Yes	Yes
02-04 LG	The OPERABILITY requirements for the source range detectors in MODE 6 are moved to the Bases.	Yes	No, see 02-01-LS21	No, see 02-01-LS21	No, see 02-01-LS21
03-01 R	The subcriticality requirement prior to irradiated fuel movement is relocated to a licensee controlled document.	Yes, See attachment 21 page 17	Yes, relocated to TRM	Yes, relocated to USAR CH 16	Yes, relocated to FSAR CH 16
04-01 LG	This change removes the word "automatic" from the requirement that each penetration be capable of being closed by an OPERABLE automatic containment purge isolation valve. The requirement for an automatic valve would be stated in the Bases.	Yes	Yes	Yes	Yes

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
04-02 LS4	This change removes SR to perform verification within 100 hours prior to the start of core alteration or movement of irradiated fuel.	Yes	Yes	Yes	Yes
04-03 LS5	The frequency of verifying that Containment ventilation isolation occurs is changed from 7 days to 18 months.	Yes	Yes	Yes	Yes
04-04 TR1	The surveillance requirement is revised to allow for increased flexibility in using an actual or simulated actuation signal. Identification of the specific actuation signal is moved to the Bases.	Yes	Yes	Yes	Yes
04-05 LG	This change moves SR to verify the trip set point concentration value for the Containment Purge Monitors is reset during CORE ALTERATIONS or other movement of irradiated fuel in containment.	No, not in CTS.	No, not in CTS	No, not in CTS	Yes, moved to FSAR
04-06 LS23	The requirement to verify the capability to close the containment ventilation isolation valves from the control room would be removed.	No, not in CTS.	Yes	No, not in CTS	No, not in CTS
04-07 LG	The specific administrative controls used to assure personnel airlock closure capability would be moved from the Limiting Condition for Operation (LCO) to the Bases.	No, not in CTS.	Yes	No, Amendment 95 did not put Admin Control on LCO	Yes
04-08 LG	This change moves the references to Heavy Loads in the Applicability and ACTION section of CTS 3.9.4 (Containment) to the FSAR.	Yes	No, not in CTS	No, not in CTS	No, not in CTS
04-09 LS14	LCO 3.9.4 would be modified to permit an approved functional equivalent of a valve or blind flange to isolate containment penetrations.	Yes	Yes	No, already in CTS (Amendment 74)	Yes
04-10 LS20	Adds a footnote stating that penetration flow paths that provide direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.9

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
05-01 R	This change relocates the CTS section dealing with maintaining direct communication between the control room and the refueling station.	Yes, see Attachment 21 page 19	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103
06-01 R	This change relocates the CTS section for the Manipulator Crane.	Yes, see Attachment 21 page 21	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103
07-01 R	This change relocates the CTS section dealing with crane travel.	Yes, see Attachment 21 page 23	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103
08-01 A	This change provides clarification that loading irradiated fuel assemblies is the activity that could increase the reactor decay heat load.	Yes	Yes	Yes	Yes
08-02 A	This change removes a limit on Reactor Coolant System temperature.	No, not in CTS.	No, not in CTS	Yes	Yes
08-03 LS6	This change allows the removal of the Residual Heat Removal (RHR) loop from operation for additional purposes other than the performance of core alterations in the vicinity of the hot legs.	Yes	Yes	Yes	Yes
08-04 A	This change eliminates the option of securing RHR prior to initial criticality since initial criticality has already occurred.	Yes	Yes	Yes	No
08-05	Not Used	N/A	N/A	N/A	N/A
08-06 M	This change adds an additional surveillance requirement to verify correct breaker alignment and indicated power available at least once per 7 days.	Yes	Yes	Yes	Yes
09-01 A	The requirements of this LCO would be incorporated into ITS 3.9.4, "Containment Penetrations" or would be addressed by ITS LCO 3.0.3 and 3.0.4.	Yes	No - CPSES does not have containment ventilation specs in CTS 3/4.9.	Yes	Yes

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-02 LS7	This change deletes the requirement to close each purge valve when the containment ventilation system is inoperable.	Yes	No, CPSES does not have containment ventilation specs in CTS 3/4.9.	Yes	Yes
09-03 LS8	The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the SR frequency from once per 7 days and within 100 hours prior to CORE ALTERATIONS to once per 18 months.	Yes	No, CPSES does not have containment ventilation specs in CTS 3/4.9.	Yes	Yes
09-04 LS15	Removes the requirement for immediate action when one containment purge monitor is inoperable. ITS LCO 3.3.6 will allow one purge monitor to be inoperable for up to 4 hours during CORE ALTERATIONS or movement of irradiated fuel in containment.	No, not in CTS.	No, CPSES does not have Containment Ventilation Spec in CTS 3/4.9	No, Plant design different	Yes
10-01 R	This change relocates the CTS requirements concerning reactor vessel water level for movement of control rods.	Yes, see Attachment 21 page 25	Yes, relocated to TRM	No, relocated per Amendment 89	No, relocated per Amendment 103
10-02 LS22	This change deletes the surveillance requirement to verify water level within 2 hours prior to the start of movement of irradiated fuel assemblies.	Yes	Yes	Yes	Yes
10-03 LS18	Revises Applicability such that it applies only when irradiated fuel is moved. The CTS also applies to movement of un-irradiated fuel when there is irradiated fuel in the core.	Yes	Yes	No, already in CTS	No, already in CTS
11-01 LG	This change modifies the Applicability to during movement of irradiated fuel assemblies in the fuel storage pool to be consistent with the Fuel Handling Accident (FHA).	Yes, moved to an ECG	Yes	Yes	Yes
11-02 LS10	This change modifies the ACTION statement and eliminates the time requirement to restore water level.	Yes	Yes	Yes	Yes

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
11-03 M	Revised to delete "irradiated fuel assemblies seated in" since the accident analysis assumes fuel assembly lying on top of the fuel storage racks.	No, licensed FHA analysis supports "assemblies seated in."	Yes	No, licensed FHA analysis supports "assemblies seated in."	Yes
11-04 LG	Moves the restriction on crane operation to a licensee controlled document.	Yes, To FSAR	Yes, Moved to TRM	Yes, to USAR	Yes, to FSAR
12-01 LS24	The Applicability would be changed to "During movement of irradiated fuel in the fuel building" instead of "Whenever irradiated fuel is in the spent fuel pool."	Yes	No, CPSES does not have this specification	Yes	Yes
12-02 LG	Removes the restriction on crane operations over the spent fuel storage areas when the fuel building air cleanup system was inoperable.	Yes, To FSAR	No, CPSES does not have this specification	Yes, to USAR	Yes, to FSAR
12-03 A	The statement that 3.0.3 [and 3.0.4] are not applicable would be removed.	No, FHBVS not applicable in MODE 1, 2, 3, and 4	No, CPSES does not have this specification	Yes	Yes
12-04 A	The Surveillance Requirements regarding filter testing would be moved to a "Ventilation Filter Testing Program" that is called out in the Administrative Controls Section 5.5.11 of the ITS.	Yes	No, CPSES does not have this specification	Yes	Yes
12-05 TR1	This change revises the SR to allow for increased flexibility in using an actual or simulated actuation signal. Identification of the specific actuation signal is moved to the Bases.	Yes	No, CPSES does not have this specification	Yes	Yes
12-06 A	This requirement would have the operability of each train of the [emergency exhaust system] (including maintaining negative pressure in the building) to be demonstrated.	Yes	No, CPSES does not have this specification	Yes	Yes
12-07 LS25	Removes 31 day STAGGERED TEST BASIS to be consistent with ITS.	No, not in CTS.	No, CPSES does not have this specification	Yes	Yes

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-08 LS16	The proposed change would allow the 18-month testing of the [FHBVs] ability to maintain the required pressure differential between the building and the outside atmosphere to be performed on a STAGGERED TEST BASIS.	Yes	No, CPSES does not have this specification	Yes	Yes
12-09 LG	The requirement for an OPERABLE emergency power source for an OPERABLE FHBV train is moved to the Bases.	Yes	No, CPSES does not have this specification.	No	No
12-10 LS9	The "within 31 days after removal" requirement for completion of laboratory analyses is deleted.	Yes	No, CPSES does not have this specification in CTS 3/4.9.	Yes	Yes
12-11 A	The SR to measure [FHBVs] flow rate is not identified as a separate SR in the ITS because it is verified during the other in-place filter tests (see ITS 5.5.11 a. and b.)	Yes	No, CPSES does not have this specification in CTS 3/4.9.	Yes	Yes
12-12 LS26	This change establishes appropriate ACTIONS and Completion Times for Fuel Building pressure envelope degradation.	{No, maintaining CTS}	No, CPSES does not have this specification in CTS 3/4.9	Yes	No, maintaining CTS
14-01 LS11	This change deletes the restrictions on placing spent fuel assemblies into Region 2 of the spent fuel pool and changing storage locations designations from Region 1 to Region 2.	No, Requirement not in CTS.	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes
14-02 M	This changes the Applicability from "Whenever irradiated fuel assemblies are in the spent fuel pool" to "Whenever any fuel assembly is in Region 2 of the spent fuel pool."	No, already in CTS.	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes
14-03 LS12	This change would delete the ACTION requirements to suspend all other movement of spent fuel and crane operations.	Yes	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.9

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
14-04 LS13	Deletes the ACTION statement requirement to verify spent fuel boron concentration every 8 hours while action is being taken to relocate noncomplying spent fuel assemblies from Region 1 to Region 2.	Yes	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes
14-05 LG	The requirement to keep records of the burnup analysis for all assemblies in Region 1 and 2 would be relocated to a licensee controlled document.	Yes, to the Bases	No, CPSES does not have this specification in CTS 3/4.9	Yes, to USAR	Yes, to FSAR
14-06	Not Used	N/A	N/A	N/A	N/A
14-07	Not Used	N/A	N/A	N/A	N/A
14-08	Not Used	N/A	N/A	N/A	N/A
14-09	Not Used	N/A	N/A	N/A	N/A
14-10 A	The statement that 3.0.4 is not applicable would be removed.	Yes	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes
15-01 R	The requirement to empty the spent fuel exclusion zone area prior to any spent fuel shipping cask handling operations is relocated to Licensee controlled document.	Yes, see Attachment 21page 27	No, CPSES does not have this specification in CTS 3/4.9	No, not in CTS	No, not in CTS

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR1, 2, 3...." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1

10CFR50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would eliminate the requirement to borate at a specified flow rate with a specified concentration of borated water until the boron concentration limit or $K_{eff} \leq 0.95$ is achieved. The boron limit is specified in the Core Operating Limit Report (COLR) and is based on maintaining the required K_{eff} .

The requirements to initiate boration at a specified flow rate having a specified boron concentration would be replaced by the more general requirement to initiate boration to restore the required boron concentration. The reactor operators are expected to select the best method of increasing the boron concentration to the required value specified in the COLR. As described in the ITS Bases, the required flow rate and boron concentration should be selected depending on plant conditions and available equipment. The ITS Bases 3.1.1 allow the operator to use the "best source available for the plant conditions." This is an example of maintaining the overall safety requirement in TS but removing procedural details from the TS allowing the plant operator the ability to select the appropriate procedure and equipment for the existing plant condition.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The ACTION statement that would be modified requires that the boron concentration limit be restored if not within specification by borating at a specific flow rate with a specific concentration of boron. The proposed change would eliminate details regarding flow rates and concentrations from the Technical Specifications; however, the overall requirement to restore the boron concentration would remain. Removal of the details of system operation would not affect the probability of an accident occurring. Also, the changes would have a negligible effect on the consequences of postulated accidents; because the changes may impact the timing of restoration of boron concentration and, as reflected in the Bases to the ITS, there is no unique Design Basis Event that must be satisfied - only that restoration of boron concentration be done as soon as possible. In addition, activities that could affect reactivity such as CORE ALTERATIONS must be suspended whenever the concentration is below the limit. The Technical Specifications do not provide a limit on the time to restore SDM. For consequences of an accident to be affected, an accident that is impacted by boron concentration would have to occur during the short time period that concentration is not within limit. The probability of that is negligible because of the short duration expected to restore boron concentration to within limits. Therefore there would be no significant increase in the probability or consequences of a previously evaluated accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes may affect the timing of restoration of the boron concentration depending on the equipment selected by the operator to perform the boration. The change involves no hardware modifications or changes in the manner in which plant systems perform their functions. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The SDM limits are not changed nor are there any changes to accident analysis assumptions, methodologies, credited protection/mitigation equipment, or event acceptance criteria. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limits, DNBR limits, F_0 , F_H , LOCA PCT, peak local power density, or any other margin of safety. Therefore, the proposed changes do not involve a significant reduction in any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS1" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2

10CFR 50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, the proposed change would delete the Surveillance Requirement (SR) to determine boron concentration and k_{eff} within limits before removing or unbolting the reactor vessel head or before withdrawing any control rod in excess of three feet from its fully inserted position. The first of these requirements is redundant to the requirement to imposed by the APPLICABILITY NOTE in ITS 3.9.1 to meet the Limiting Condition for Operation (LCO) prior to entering MODE 6 from MODE 5. Compliance with the LCO is assured by verifying boron concentration in accordance with ITS SR 3.9.1.1. In this case, unbolting the vessel head in preparation for removal is part of the definition of MODE 6. Therefore, this requirement is redundant to the requirement to verify boron concentration prior to entry into MODE 6.

The requirement that involves withdrawal of control rods is also redundant because the analysis used to determine the boron concentration limit specified in the COLR considers the most adverse conditions of fuel assembly and control rod position. The boron concentration is sufficient to maintain $k_{eff} \leq 0.95$ with the most reactive rod control cluster assembly completely removed from its fuel assembly.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The removal of this SR would have an insignificant effect on the probability of occurrence of a criticality event because the boron concentration limit that is required for the LCO would be in effect at the time that the vessel head is unbolting or removed; and the analysis considers the various conditions of reactor control rods that may be allowed by plant procedures. Therefore, the probability of an accident would not be significantly increased by the proposed removal of the SR.

The SR in question deals with performance of activities intended to prevent an accident from occurring by establishing plant conditions to prevent a criticality event. In view of the requirement to meet boron concentration limits prior to entry into MODE 6, the nonperformance of the SR would have no effect on the consequences of an accident.

Therefore, the proposed change would have no significant effect on the probability or consequences of any previously analyzed accidents.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS2 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The SR that is proposed for elimination would require verification of boron concentration just prior to performing specific evolutions, i.e., detensioning of reactor vessel head bolts and CORE ALTERATIONS involving control rods. Eliminating the SR would not affect the manner of operation of the plant or any plant systems so that the occurrence of a new kind of accident could result. This proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The proposed elimination of the SR would have the effect of relying on alternate means to maintain the margins for preventing a criticality event. Based on the previous discussion, the requirement regarding boron concentration when entering MODE 6 and the analytical assumptions used to establish the boron concentration limit provide assurance that the margins would be maintained. Therefore, the proposed change would have no significant adverse effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS2" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, the requirements of SR 4.9.2 b. and c. to perform ANALOG CHANNEL OPERATIONAL TESTS (ACOTs) would be deleted and a SR to perform a Channel Calibration is added, in accordance with NUREG-1431. In MODE 6, the source range monitors are required for indication only and there are no precise setpoints associated with these instruments. In this capacity, the source range instrumentation is typically used to read a relative change in count rate. The source range instrumentation is monitored for significant changes in count rate which are important to evaluate the change in core status. The accepted convention defining criticality does not require precise or specific set points or indication, but only requires verification of a slowly increasing count rate. Consistent with NUREG-1431, indicating instruments only require channel checks and channel calibrations. The more frequent ACOTs are applied only to those channels with operational interlocks or other setpoint actuations. Therefore, the MODE 6 channel checks and channel calibration requirements for the source range monitors are adequate to assure their operability considering the accuracy required of these instruments in MODE 6.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

During REFUELING, the source range monitors provide visual and audible indication of neutron count rate to plant operators. The proposed deletion of ACOTs for these channels would be offset by the CHANNEL CHECK and CHANNEL CALIBRATION requirements. The addition of a CHANNEL CALIBRATION to be performed every 18 months provides assurance that the instruments can provide the visual and audible indication. There are no alarms, interlocks, or trip setpoints associated with these channels that are required to be OPERABLE during MODE 6. Thus, the proposed change would have no effect on the probability of an accident occurring. In addition, in Mode 6 the source range instruments provide no automatic actuation function used for mitigation of accidents, and they would have no effect on the outcome of an accident. Furthermore, the modification of SRs for these indicators does not imply that they will be unavailable when required. The CHANNEL CHECKS and CHANNEL CALIBRATION SR that remain in effect provide the necessary assurance of OPERABILITY. Therefore, there would be no increase in the probability or consequences of a previously evaluated accident as a result of making the proposed change.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change would not create the possibility of a new or different accident, because the plant or its systems would not be operated any differently. The proposed change has to do with the type of SR applied to source range instrument channels. Therefore, there would be no operational changes to contribute to the possibility of a new accident resulting from the proposed change. Therefore, this change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The margins of safety in question are those involved with preventing criticality during REFUELING operations. The monitors provide visual and audible indication of neutron count rate, and, therefore, provide assurance that the core reactivity is being maintained. However, reactivity is maintained primarily by the requirements of ITS 3.9.1 which assure that the boron concentration in refueling water is within limit. Thus the neutron monitoring channels provide further assurance that criticality will not occur. In addition, deletion of certain channel tests (ACOTs) would not prevent the channels from performing when required. The CHANNEL CHECK and CHANNEL CALIBRATION SRs for this equipment are appropriate for the function of the source range channels during REFUELING. Therefore, the deletion of ACOTs for the source range neutron monitoring channels during MODE 6 would have an insignificant effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS3" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would remove the Surveillance Requirement (SR) to perform verification that the containment penetration Limiting Condition for Operation (LCO) was met within 100 hours prior to the start of core alteration or movement of irradiated fuel.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The purpose of the SR is to assure the OPERABILITY of the containment penetrations that must be closed or capable of closing to prevent the release of radioactivity in the event of a Fuel Handling Accident. So, the SR is intended to assure that mitigation features are available and has no impact on the probability of an accident occurring. The Applicability statement for this LCO is "During CORE ALTERATIONS or movement of irradiated fuel within the containment." Thus the requirement to verify the LCO is met within 100 hours of starting the evolutions for which the LCO is applicable is redundant; because the LCO must be met at the time that the evolutions occur. Only the timing (7 days versus 100 hours) is different. Therefore, the proposed change would involve no increase in either the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change revises the timing for performance of the SR. Changes in SR frequency would not lead to changes in plant or plant system operations or other conditions that could cause an accident of a new or different type. Thus, the proposed change does not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

As noted in the evaluation of Criterion 1 above, the requirement to verify the LCO is met within 100 hours of starting the evolutions for which the LCO is applicable is redundant; because the LCO must be met at the time that the evolutions occur. In addition, the SR requires that the verification of containment isolation capability be performed once every 7 days. This requirement would remain in the ITS. From the standpoint of equipment OPERABILITY, there is little difference between performing the surveillance within 100 hours or within 168 hours of beginning the applicable activities. Thus, removing the requirement would have little effect on the availability of accident mitigating equipment. Therefore, the proposed changes would result in an insignificant adverse effect on any margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS4" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change to the Surveillance Requirement (SR) for containment building penetrations during CORE ALTERATIONS or moving irradiated fuel in containment would revise the frequency of verifying that containment ventilation isolation occurs from 7 days to 18 months. This is in accordance with NUREG-1431.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The change involves a revision in the frequency of testing accident (Fuel Handling Accident, FHA) mitigating equipment. This would have no effect on the probability of previously analyzed accidents. The proposed change to 18 months would apply the same frequency of testing to containment purge isolation valves as applied to other containment isolation valves that must be OPERABLE during reactor operations. The 18-month frequency has been found adequate for the type of testing applied to instrumentation and valves that must mitigate events much more severe and much more challenging to the containment boundary (e.g., LOCA, MSLB) than the FHA. Therefore, the proposed change would have an insignificant effect on the consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change would not create the possibility of a new or different accident, because the plant or its systems would not be operated any differently. The proposed change affects the frequency of testing accident mitigating equipment. Therefore, there would be no operational changes to contribute to the possibility of a new accident resulting from the proposed change. Therefore, this change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS5 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The margins of safety in question are those involved with radiological consequences of a FHA. As noted in the evaluation of Criterion 1 above, the proposed change in frequency for purge isolation valve testing would revise the testing to be in line with the test requirements for other containment isolation valves. Therefore, the proposed change would not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS5" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with a Note to LCO 3.9.5 of NUREG-1431, this change allows the removal of the residual heat removal (RHR) loop from operation for additional purposes other than the performance of core alternations in the vicinity of the hot legs. This allows increased flexibility for operations such as core mapping and isolation valve testing. No operations are permitted that would cause a reduction of the reactor coolant system (RCS) boron concentration. RHR flow is used for decay heat removal and for boron mixing during MODE 6 operations.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change would permit the securing of RHR flow through the reactor vessel for up to one hour in every eight hours provided no operations involving a reduction in boron concentration were performed. The two RHR functions are adequately assured during operation in accordance with this note; because, decay heat removal will be accomplished by the natural circulation of the large amount of water in and above the RCS (≥ 23 feet above the reactor vessel flange), and boron mixing concerns are avoided by prohibiting evolutions that would reduce the boron concentration. The one-out-of-eight hour restriction has been shown to be adequate for decay heat removal. Continued decay heat removal assures that the reactor remains in a safely cooled condition, and adequate boron mixing assures that an inadvertent criticality would not occur. Although the change could increase the number of times that RHR flow may be secured, the restrictions discussed above would be maintained such that the proposed change would not involve a significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change would allow RHR flow to the reactor vessel to be stopped for short periods of time. The duration has been selected to assure continued adequate fuel cooling and criticality prevention. The CTS already permit this interruption in RHR operation but under more limited conditions. The proposed change in the frequency of performing this evolution would not lead to changes in plant or plant system operations or other conditions that could cause an accident of a new or different type. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

As noted above, the one-out-of-eight hour securing of RHR flow has been shown to be acceptable for decay heat removal. Also, the issue of boron mixing is addressed by a restriction on evolutions that may reduce the boron concentration. Therefore, the proposed change would not result in any decrease in the margin of safety defined in the accident and other analyses that depend on decay heat removal. Thus, the proposed change would not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS6" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would delete the requirement to close each purge valve when the Containment Ventilation System is inoperable. The improved Technical Specification (ITS) only requires that core alterations and irradiated fuel movement be suspended. The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the actions required when the ventilation system is inoperable from closing the purge valves to suspending core alterations and irradiated fuel movement.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The applicability of the Limiting Condition for Operation (LCO) and required actions for both the current Technical Specification (CTS) and ITS 3.9.4 are identical, i.e., During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Therefore, neither of these LCOs would be in effect if CORE ALTERATIONS or movement of irradiated fuel were suspended. The function of the purge valves is to close following a fuel handling accident (FHA) to prevent the escape of radioactivity from containment. Therefore, the proposed changes would have no effect on the probability of a FHA. In addition, the change from requiring the valves to be closed to prevent radioactivity release to suspending activities which could lead to a FHA (and radioactivity release) would have the same effect with regard to consequences of the accident. Therefore, the proposed change to the purge valve action statement would have an insignificant effect on the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change in the purge isolation valve action statement would not lead to changes in plant or plant system operations or other conditions that could cause an accident of a new or different type. The proposed change, therefore, would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS7 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

As noted in the evaluation of Criterion 1 above, the proposed action would prevent the occurrence of a FHA and would be preferable to the current action which aligns the purge valves in preparation for mitigating a FHA. Both actions have similar intent with regard to preventing releases of radioactivity. However, the former provides increased margins of safety by avoiding the occurrence of the accident in the first place. Therefore, the proposed change would not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS7" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would remove the SR to test purge isolation manually and with each channel of high radiation test signals within 100 hours of starting CORE ALTERATIONS and every 7 days during CORE ALTERATIONS. The containment ventilation TS requirements would be integrated into ITS 3.9.4. This has the effect of changing the SR frequency from once per 7 days and within 100 hours prior to CORE ALTERATIONS to once per 18 months.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The purpose of the SR is to assure the OPERABILITY of the containment penetrations that must be closed or capable of closing to prevent the release of radioactivity in the event of a Fuel Handling Accident. So, the SR is intended to assure that mitigation features are available and has no impact on the probability of an accident occurring. In addition, the change involves a revision in the frequency of testing accident (Fuel Handling Accident, FHA) mitigating equipment. This would have no effect on the probability of previously analyzed accidents.

The Applicability statement for this Limiting Condition for Operation (LCO) is "During CORE ALTERATIONS or movement of irradiated fuel within the containment." Thus the requirement to verify the LCO is met within 100 hours of starting the evolutions for which the LCO is applicable is redundant; because the LCO must be met at the time that the evolutions occur. Also, the proposed change to 18 months would apply the same frequency of testing to containment purge isolation valves as applied to other containment isolation valves that must be OPERABLE during reactor operations. The 18-month frequency has been found adequate for the type of testing applied to instrumentation and valves that must mitigate events much more severe and much more challenging to the containment boundary (e.g., LOCA, MSLB) than the FHA. Therefore, the proposed change would have an insignificant effect on the consequences of a previously evaluated accident. Based on the previous discussion, the proposed change would not involve an increase in the probability or consequences of a previously evaluated accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS8 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change affects the frequency of performing a SR for purge isolation; it would not alter plant hardware or the manner in which systems and equipment perform their safety function. Therefore, the change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The margins of safety in question are those involved with radiological consequences of a FHA. As noted in the evaluation of Criterion 1 above, the requirement to verify the LCO is met within 100 hours of starting the evolutions for which the LCO is applicable is redundant because the LCO must be met at the time that the evolutions occur. In addition, the SR requires that the verification of containment isolation capability be performed once every 7 days. This requirement would remain in the ITS. From the standpoint of equipment OPERABILITY, there is little difference between performing the surveillance within 100 hours or within 168 hours of beginning the applicable activities. Thus, removing the requirement would have little effect on the availability of accident mitigating equipment. As noted in the evaluation of Criterion 1 above, the proposed change in frequency for purge isolation valve testing would revise the testing to be in line with the test requirements for other containment isolation valves. Therefore, the proposed change would not involve a significant reduction in any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS8" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The Applicability is revised to not require the boron concentration to be greater than 2000 ppm unless fuel in the pool has been moved and a position verification has not been performed. The boron concentration is required to prevent inadvertent criticality due to a mis-positioned high enrichment fuel assembly. If a pool verification has confirmed that the fuel bundles are in the correct positions, no inadvertent criticality can occur due to the design of the racks. The ITS further require suspension of fuel movement and either restoration of the pool boron concentration or the verification of storage of the high enrichment fuel in the proper pool location.

NO SIGNIFICANT HAZARDS

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes do not alter the plant configuration, operation or function of any safety system. Consequently the changes do not significantly increase the probability of an accident as defined in the FSAR. Maintenance of the fuel pool boron concentration is an accident prevention feature, i.e., criticality, and is only required to be maintained if there is a potential for a mispositioned high enrichment fuel assembly. If the fuel positions have been verified the storage racks are designed to prevent criticality.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not require physical alteration to any plant system or change the method by which any safety-related system performs its function.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS9 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed changes do not alter any basic regulatory requirements or change any accident analysis assumptions, initial conditions or results. Consequently, the proposed changes do not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based upon the preceding information, it has been determined that the proposed changes associated with NSHC "LS9" do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in margin of safety. Therefore, it is concluded that the proposed change satisfy the no significant hazards consideration standards of 10CFR 50.92© and, accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change would modify the ACTION statement to eliminate the time requirement to restore water level.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves actions to take during fuel movement if the required water level is not maintained. The deletion of a four hour limit to restore water level is acceptable because the ACTION statement would require the suspension of movement of fuel assemblies which would remove the applicability of the Limiting Condition for Operation (LCO) and make the water level requirement not applicable. Therefore, the proposed change would not result in a significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves removing the four hour limit for regaining spent fuel pool level. Eliminating this requirement would not create the possibility of a new or different accident because the four hour limit relates to timing of completing a corrective action. Thus, the change do not involve new equipment or new methods of operating existing equipment. Therefore, the proposed change would not create the possibility of a new or different accident.

3. Does this change involve a significant reduction in a margin of safety?

The removal of the four hour time limit to restore water level has no effect on safety margins because the action of suspending fuel movement renders the time limit redundant. Therefore, the proposed changes do not involve a reduction in a margin of safety.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS10
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS10" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS12
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, this change would delete the action requirement to suspend all other movement of spent fuel and crane operations if the storage requirements of Region 2 are not met.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The concern of this TS is criticality in the spent fuel pool. Deleting the actions pertaining to crane operations and movement of other spent fuel (other than the fuel erroneously loaded into Region 2) would focus the actions on moving the mis-located spent fuel assembly back to Region 1. The required action of the ITS requires immediate action to move the noncomplying fuel from Region 2. Deleting the crane operation and other spent fuel handling would eliminate extraneous actions that have little to do with criticality prevention. Therefore, based on the foregoing, the proposed change would not result in a significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes involve no changes to the plant design or changes in operation of plant systems. Therefore, this proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS12 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The margin of safety are maintained in this case by moving the noncomplying fuel from Region 2. The requirements on crane operation and other fuel movement are incidental to the movement of the noncomplying fuel. Therefore, no reduction in a margin of safety would result from this proposed change.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS12" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS13
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, the proposed change would delete the ACTION statement requirement to verify spent fuel boron concentration every 8 hours while action is being taken to relocate noncomplying spent fuel assemblies from Region 2 to Region 1. The Frequency of verification of boron concentration would be in accordance with Improved Technical Specification (ITS) 3.7.16, "Fuel Storage Pool Boron Concentration," i.e., every 7 days.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The requirement to verify boron concentration in the spent fuel pool once per 8 hours is intended to prevent an inadvertent criticality due to an accident occurring while noncomplying fuel is present in Region 2. A criticality event resulting from an accident would be prevented by maintaining the boron concentration ≥ 2000 ppm. Deletion of this requirement would have an insignificant effect because the boron concentration in the spent fuel pool changes very slowly over time. The concentration of boron in the spent fuel pool is maintained at ≥ 2000 ppm in accordance with ITS 3.7.16, "Fuel Storage Pool Boron Concentration." ITS SR 3.7.16.1 requires verification of boron concentration once per 7 days because, as stated in the Bases, no major replenishment of pool water is expected to take place over such a short period of time. Thus, if noncomplying fuel assemblies are inadvertently placed in Region 2, the Required Action is to move the assemblies. Since the fuel pool boron concentration does not change rapidly over time, there is no need to verify the concentration more frequently while the movement of fuel from Region 2 is occurring. Therefore, the proposed change would not result in a significant increase in the probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves removing a surveillance requirement on spent fuel pool boron concentration. The intent of the surveillance is to prevent a criticality accident. The proposed change is not related to plant design changes, equipment modifications, or to changes in system operation. Therefore, this proposed change would not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS13 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The requirement to verify boron concentration every 8 hours has little effect on the potential effects of an accident that occurred while the provisions of LCO 3.7.16 were not met. Boron concentration in the fuel pool does not change rapidly over time, and the focus of operations is to quickly remove the noncomplying fuel from Region 2 and, thus, meet the requirements of the LCO. Therefore, removing this action requirement would not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS13" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Limiting Condition for Operation (LCO) 3.9.4 would be modified to permit an approved functional equivalent of a valve or blind flange to isolate containment penetrations.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change to containment building penetrations during CORE ALTERATIONS or movement of irradiated fuel in the containment is identical to changes previously approved by the NRC for other nuclear plants. During plant shutdown for refueling operations, a fuel handling accident is the limiting event postulated for the design of containment accident mitigation features. A fuel handling accident assumes that a spent fuel assembly is dropped and the fuel rods rupture. This event does not pressurize containment such that the containment structure must perform as a pressure boundary to prevent the release of radioactivity. The NRC has stated that, "Given that a fuel handling accident is unlikely to result in pressurization of the containment, the closure of potential release paths by other means would also limit the consequences of such accidents to below the criteria given in Section 15.7 of the Standard Review Plan. This position was adopted by the staff in NUREG-1431, Standard Technical Specifications: Westinghouse plants. The NRC further concluded that the use of a temporary closure device in lieu of a closed isolation valve or blind flange provides an effective means of preventing the release of radioactive material following a fuel handling accident provided the temporary devices are designed, fabricated, installed, tested, and utilized in accordance with established procedures. The Bases of Improved Technical Specification (ITS) 3.9.4, "Containment Penetrations" (NUREG-1431) endorse the use of "equivalent isolation methods."

The use of isolation devices as penetration closure would not cause an accident to occur; these devices would perform an accident mitigating function. Therefore, the proposed TS change would not affect the probability of an accident. Also, the conclusions reached by the NRC staff, as evidenced by the Bases to the ITS, demonstrate that the proposed use of these functionally equivalent devices would not significantly increase the consequences of a fuel handling accident. Therefore, the proposed change would not result in a significant increase in the probability or consequences of a previously evaluated accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS14 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change would implement an alternate means of containment isolation suitable for mitigating a fuel handling accident. The change would not involve design or operating changes that could cause a new accident to occur. Therefore, this proposed change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

As discussed in the evaluation of Criterion 1 above, the event of concern during CORE ALTERATIONS or movement of irradiated fuel in containment is the fuel handling accident in containment. The NRC previously has approved TS changes based on the conclusion that the use of functionally equivalent closure devices, in a controlled manner, provides an acceptable means of mitigating these accidents. Thus, the proposal would maintain the level of safety equivalent to the original accident mitigating features. Therefore, the proposed changes would involve no significant reduction in the margins of safety for the plant.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS14" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16 10CFR 50.92 EVALUATION FOR TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would allow the 18-month testing of the FHBVS ability to maintain the required pressure differential between the building and the outside atmosphere to be performed on a STAGGERED TEST BASIS in accordance with NUREG-1431. The current requirement Frequency is simply 18 months. The new STS define STAGGERED TEST BASIS such that one train of the system would be tested every 36 months rather than the 18 months required in the current Specification. This is a relaxation in testing requirements.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The FHBVS is an accident mitigation system; it is not a precursor to any previously analyzed accident. Therefore, the proposed change to the frequency of testing would not affect the probability of an accident occurring. The proposed change would reduce the frequency of testing the building pressure reduction feature of the system. However, the system would still be tested for operability by (1) monthly operation, (2) 18 month automatic start, and (3) testing under the Ventilation Filter Test Program. Also, the factors that affect building pressurization, in addition to the capability of the fan, do not change significantly over time and intentional changes to them are performed under administrative controls. These factors include alternations to the building pressure envelope (piping and electrical penetrations, and doors). Since these are controlled (a retest of the exhaust system would be required if they changed significantly) and since the fan operability is checked monthly, the proposed change would still provide assurance that the FHBVS remains OPERABLE. Also, primary degradation of the exhaust fans normally occurs over an extended period. Operating experience based on maintaining similar equipment further supports the acceptability of the proposed test interval. Thus, the proposed reduced frequency of testing does not significantly increase the consequences of an accident. Therefore, the proposed change would have an insignificant effect on the consequences of a previously evaluated accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change would not create the possibility of a new or different accident, because the proposed change would not involve new equipment, or operating existing equipment in a new manner. The FHBVS is not an accident initiator. And the proposed change would still assure that the system could perform its required function. Therefore, this change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The margins of safety in question are those involved with mitigating radiological consequences of accidents. As noted above, the increased interval between pressurization operability testing is acceptable based on the other testing done to assure the OPERABILITY of the system and the controls placed on maintenance of the building pressure envelope. Therefore, the proposed change would not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS16" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS18
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, this change revises applicability such that it applies only when irradiated fuel is moved. The CTS also applies to movement of unirradiated fuel when there is irradiated fuel in the core. The LCO applies to movement of irradiated fuel to ensure that a sufficient water level exists above the reactor vessel flange to retain iodine fission products in the event that the transported irradiated fuel assembly is dropped and releases its fission products. The release of fission products from irradiated fuel that is in the reactor core, as a result of damage caused by a dropped unirradiated fuel assembly, is considered to be highly unlikely.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change removes the Applicability of the LCO during movement of unirradiated fuel assemblies when the core contains irradiated fuel assemblies. Due to the configuration of the fuel support assembly within the core (which provides protection to the contained fuel assemblies), the probability of a significant fission product release from the core as a result of dropping an unirradiated fuel assembly on the core, is small. The CTS only addresses the movement of fuel assemblies. Heavy loads are addressed in the FSAR by controlling the size of the load and the method of lifting the load to minimize the probability that a load is dropped. In this regard a new fuel assembly is a heavy load and the consequences of a dropped un-irradiated fuel assembly would be no more than another load of similar weight and shape. Therefore this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change removes the Applicability of the LCO during movement of unirradiated fuel assemblies when the core contains irradiated fuel assemblies. Eliminating this applicability would not create the possibility of a new or different accident because the analysis of a dropped irradiated fuel assembly is still the limiting accident of this type. The changes do not involve new equipment or new methods of operating existing equipment. Therefore, the proposed changes would not create the possibility of a new or different accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS18 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change has no affect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS18" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS20
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A note is added to Limiting Condition for Operation (LCO) 3.9.4.c and to the 7-day SR to state that containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere may be open under administrative controls. The note would allow these penetrations to be unisolated during CORE ALTERATIONS and movement of irradiated fuel assemblies within containment provided that specified administrative controls were employed. This note is consistent with proposed traveler WOG-76. The proposed Note is acceptable based on administrative controls that consist of written procedures that require designated personnel having knowledge of the open status of the valves in question and specified persons designated and readily available to isolate the open penetration in the event of a fuel handling accident. These administrative controls are used to establish containment closure for a containment personnel airlock. The NRC staff has allowed changes to the requirements for airlocks that allow both doors of an airlock to be open during CORE ALTERATIONS and during movement of irradiated fuel inside containment provided that administrative controls are in place to quickly close one door and establish containment closure.

This proposed Technical Specification (TS) change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves changes to the TS requirements for containment closure which is an accident mitigating feature. The changes would not affect the likelihood of occurrence of any accident previously evaluated. The proposed change does not involve any hardware or plant design changes. The containment leakage value is not assumed to be an initiator of any analyzed event. The isolation valve, or temporary closure device, serves to limit the consequences of accidents. The proposed change would ensure the isolation valves, or functional equivalent, will perform their required containment closure function and will serve to limit the consequences of a fuel handling accident as described in the [FSAR] such that the results of the analyses in the [FSAR] remain bounding. In considering the consequences of a design basis fuel handling accident inside containment, the assumptions in the analysis take no credit for containment isolation as a barrier to prevent the postulated release of radioactivity. For events that would occur during CORE ALTERATIONS or movement of irradiated fuel assemblies, containment closure is considered a defense-in-depth boundary to prevent uncontrolled release of radioactivity. Additionally, the proposed change does not impose any new safety analyses limits or alter the plant's ability to detect and mitigate events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS20 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change involves reliance on manual actuation of containment penetration valves or closure devices rather than automatic systems or passive devices (blind flanges or closed valves) to block the unimpeded flow of the containment atmosphere to the environs. The proposed change would not necessitate a physical alteration of the plant features that provide core cooling or subcriticality (no new or different type of equipment will be installed) or changes in parameters governing plant operation during CORE ALTERATIONS or movement of irradiated fuel in containment. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change is similar to the use of administrative controls to isolate an open containment airlock door. The use of administrative controls in this manner has been approved by the NRC staff for plant operations that would not require the containment to maintain a pressure boundary. This scenario is applicable during plant shutdown for refueling when CORE ALTERATIONS and movement of irradiated fuel assemblies in the containment occur. The accidental damage to the spent fuel during these operation are classified as fuel handling accidents. The proposed change allows for protection equivalent to that provided by previously approved methods of containment closure. Considering the probability of an event that would challenge the containment boundary, the alternative protection provided by this change, and the operational requirements to occasionally open these penetrations, the proposed change is acceptable and any reduction in the margin of safety insignificant.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS20" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS21
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431 and TSTF-23, Rev. 2, the requirements related to indication provided by the source range detectors would be deleted from the Limiting Condition for Operation (LCO). In accordance with TSTF-23, Rev. 2, the requirements for visual indication for plants that do not rely on a boron dilution analysis would be discussed in the Bases; while the requirements for audible indication would be eliminated as a Technical Specification requirement. In MODE 6, the source range monitors are required for indication only and there are no precise setpoints associated with these instruments. In this capacity, the source range instrumentation is typically used to read a relative change in count rate. The source range instrumentation is monitored for significant changes in count rate which are important to evaluate the change in core status. The accepted convention for defining criticality does not require precise or specific setpoints or indication, but only requires verification of a slowly increasing count rate.

Consistent with NUREG-1431, Rev. 1, the Technical Specification requirements consist of maintaining two source range neutron flux monitors OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated;
or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated;
or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

During REFUELING, the source range monitors are designed to provide visual and audible indication of neutron count rate to plant operators. The proposed move of audible indication for these channels to the Bases would not affect the availability of visual or audible indication. There are no alarms, interlocks, or trip setpoints associated with these channels that are required to be OPERABLE during MODE 6. Thus, the proposed change would have no significant effect on the probability of an accident occurring. In addition, in MODE 6 the source range instruments provide no automatic actuation function used for mitigation of accidents, and the change would have no effect on the outcome of an accident. Therefore, there would be no significant increase in the probability or consequences of a previously evaluated accident as a result of making the proposed change.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS21 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change would not create the possibility of a new or different accident, because the plant or its systems would not be operated any differently. Visual indication in the trend of reactivity would remain available to the operating staff. Therefore, there would be no operational changes to contribute to the possibility of a new accident resulting from the proposed change. Therefore, this change would not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

The margins of safety in question are those involved with preventing criticality during REFUELING operations. The monitors provide visual indication of neutron count rate, and, therefore, provide assurance that the core reactivity is being maintained. However, reactivity is maintained primarily by the requirements of ITS 3.9.1 which assure that the boron concentration in refueling water is within limit and that dilution of the boron will not occur. Thus the neutron monitoring channels provide further assurance that criticality will not occur. Therefore, moving of audible indication for the source range neutron monitoring channels to the Bases during MODE 6 would have an insignificant effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS21" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS22
10CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would remove the SR to perform verification of refueling pool level within 2 hours prior to movement of irradiated fuel assemblies

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The purpose of the SR is assure the refueling pool level is consistent with the fuel handling accident analysis assumptions. The removal of the requirement to verify refueling pool level within 2 hours of irradiated fuel movement does not impact the probability of a fuel handling accident. The consequences of a fuel handling accident are also unaffected because the LCO must still be met when irradiated fuel movement occurs. The SR for level verification within 2 hours prior to irradiated fuel movement is redundant because the SR for verifying refueling pool level every 24 hours is retained in the TS. Therefore, the proposed change would involve no increase in either probability or consequences of a previously evaluated accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change revises a frequency for performance of an SR. Changes in SR frequency would not lead to changes in the plant or plant system operation or other conditions that could cause an accident of a new or different type. Thus, the proposed change does not create the possibility of a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS22 (Continued)

3. Does this change involve a significant reduction in a margin of safety?

The requirement to verify the LCO is met within 2 hours of stating the evolutions for which the LCO is applicable is redundant; because the LCO must be met at the time the evolution occurs. In addition, the SR requires that the verification of refueling pool level be performed once every 24 hours. This requirement would remain within the ITS. Therefore, the proposed change would not result in a significant reduction in any margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS22" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS*

NSHC LS24

10CFR 50.92 EVALUATION FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE

REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The Applicability for the [Fuel Handling Building Ventilation System (FHBVS)] in CTS 3.9.12 is revised to read "during movement of irradiated fuel in the fuel handling building." The CTS Applicability, whenever irradiated fuel is in the fuel storage pool, is overly restrictive since the FHBVS function (regarding fuel building operations) is to mitigate the consequences of a fuel handling accident. The potential for a fuel handling accident exists only when fuel is being moved. The FHBVS could also provide a mitigating function in the event the spent fuel pool were excessively drained; however, that scenario is not credible given the design provisions to ensure spent fuel pool integrity and the monitoring system provided to measure pool level and annunciate low level, as described in the [FSAR Sections 9.1.2 and 9.1.3].

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the Applicability for the [FHBVS]. The proposed change in the Applicability will not affect any of the analysis assumptions for any of the accident previously evaluated. The only accident analysis crediting the [FHBVS] related to fuel building operations is the fuel handling accident. The revised Applicability ensures the system's availability for that accident by virtue of the operability requirements imposed any time irradiated fuel is being moved in the fuel handling building. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS24 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The change in Applicability will not impact the normal method of plant operation. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. Therefore, the proposed does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not affect the acceptance criteria for any analyzed event. Offsite doses in the event of a fuel handling accident will not be affected. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on any margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS24" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC TR1
10CFR 50.92 EVALUATION FOR
RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the CTS to additionally allow the use of actual actuation signals for surveillances that currently call for testing using simulated test signals only. This change achieves consistency with the proposed improved Standard Technical Specifications (ITS) (NUREG-1431).

In several specifications throughout the TS, operability of certain equipment is demonstrated by ensuring that the equipment performs its safety function upon receipt of a simulated test signal. The intent of a 'simulated' signal was to be able to perform the required testing without the occurrence (or without causing) an actual signal generating event. However, the unintended effect was to require the performance of the surveillance (using a test signal) even if an actual signal had previously verified the operation of the equipment. This change allows credit to be taken for actual events when the required equipment actuates successfully.

While the occurrence of events that cause actuation of accident mitigation equipment is undesirable, the actuation of mitigation equipment on an actual signal is a better demonstration of its operability than an actuation using a test signal. Thus the change does not reduce the reliability of the equipment tested. The change also improves plant safety by reducing the amount of time the equipment is taken out of service for testing and thereby increasing its availability during an actual event and by reducing the wear of the equipment caused by unnecessary testing.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change allows the use of an actual actuation signal (when/if it occurs) to satisfy surveillance requirements currently requiring simulated test signals to demonstrate equipment operability. While the change takes advantage of events that may have occurred, it has no adverse effect on any accident initiators or accident consequences. In fact, by potentially reducing unnecessary testing, it may reduce the probability of an accident because the testing itself can increase the probability of an accident. It may also reduce accident consequences by increasing the equipment availability (i.e., less time in test). Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

TR1 (Continued)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The use of an actual actuation signal to satisfy a surveillance requirement has either no impact on, or increases the margin of plant safety by:

- a) Increasing mitigation equipment availability, and,
- b) Improving mitigation equipment reliability by potentially reducing wear caused by unnecessary testing.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "TR1" resulting from the conversion to the ITS format satisfy the no significant hazards consideration standards of 10CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS'

Applicable Industry Travelers (1 Page)

NUREG-1431 Specifications that are not applicable (1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
3.9-1	3.9-1
3.9.2	3.9-2
3.9.3	3.9-3
3.9.4	3.9-5
3.9.5	3.9-7
3.9.6	3.9-9
3.9.7	3.9-11

Methodology (2 Pages)

Industry Travelers Applicable to Section 3.9

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-20	Incorporated	3.9-2	
TSTF-21, Rev. 1	Incorporated	None	Change made to Bases for 3.9.6
TSTF-22	Not incorporated	N/A	Changes not applicable for the specific plant application.
TSTF-23, Rev. 2	Incorporated	3.9-3	Traveler bracketed ITS 3.9.2 and revised the Bases for ITS 3.9.3 (DCPP maintaining CTS).
TSTF-51	Not incorporated	N/A	Minimal impact on plant specific applications.
TSTF-68	Not incorporated	N/A	Similar changes were incorporated into the ITS based on current licensing basis. See change description 3.9-1. (Not Appl to DCPD)
TSTF-92	Not incorporated	N/A	The proposed changes did not significantly affect current surveillance practices to warrant inclusion.
TSTF-96	Incorporated	3.9-4	
WOG-63	Not incorporated	N/A	
WOG-76	Incorporated	3.9-11	Containment penetrations allowed to be open under administrative control.

NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

Specification

Specification Title

Comments

3.9.2

Unborated Water source Isolation Vales

DCPP maintaining CTS which is based upon Licensed Dilution Accident

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

NOTE
While this LCO is not met, entry into MODE 6 from MODE 5 is not permitted.

3.9-14

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS. <u>AND</u>	Immediately
	A.2 Suspend positive reactivity additions. <u>AND</u>	Immediately
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.1.1 Verify boron concentration is within the limit specified in COLR.	72 hours

3.9 REFUELING OPERATIONS

~~(Not Used)~~

3.9-13

~~3.9.2 Unborated Water Source Isolation Valves~~

~~LC0 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.~~

APPLICABILITY: MODE 6.

ACTIONS

NOTE

~~Separate Condition entry is allowed for each unborated water source isolation valve.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. NOTE Required Action A.3 must be completed whenever Condition A is entered. One or more valves not secured in closed position.	A.1 Suspend CORE ALTERATIONS. AND	Immediately
	A.2 Initiate actions to secure valve in closed position. AND	Immediately
	A.3 Perform SR 3.9.1.1.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.2.1 Verify each valve that isolates unborated water sources is secured in the closed position.	31 hours

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One <u>required</u> source range neutron flux monitor inoperable.</p>	<p>A.1 Suspend CORE ALTERATIONS except for latching control rod drive shafts and friction testing of individual control rods.</p> <p><u>AND</u></p>	<p>Immediately <u>3.9-5</u></p> <p><u>B</u></p>
	<p>A.2 Suspend positive reactivity additions except for latching control rod drive shafts and friction testing of individual control rods.</p>	<p>Immediately <u>3.9-5</u></p>
<p>B. Two <u>required</u> source range neutron flux monitors inoperable.</p>	<p>B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.</p> <p><u>AND</u></p>	<p>Immediately <u>B</u></p>
	<p>B.2 Perform SR 3.9.1.1.</p>	<p>4 hours <u>3.9-4</u></p> <p><u>AND</u></p> <p>Once per 12 hours thereafter</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.3.2	<p>-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. -----</p> Perform CHANNEL CALIBRATION.	18 months <u>B</u>

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LC0 3.9.4 The containment penetrations shall be in the following status:

- a. The equipment hatch closed and held in place by four bolts; B
- b. One door in each air lock closed; and
- c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:
 - 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
 - 2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System valve. 3.9-7

NOTE

Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls. 3.9-11

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. <u>AND</u>	Immediately
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.9.4.1	Verify each required containment penetration is in the required status except for containment penetrations that are open under administrative controls.	7 days	<u>3.9-11</u>
SR 3.9.4.2	Verify each required containment purge and exhaust ventilation isolation valves actuates to the isolation position on an actual or simulated actuation signal.	18 months	<u>B-PS</u>

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation-High Water Level

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

-----NOTE-----
The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

~~The required RHR loop may be removed from operation for ≤ 2 hours per 8 hour period for performance of leak testing the RHR suction isolation valves provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.~~ 3.9-8

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	<u>AND</u>	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>	
	A.4 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.9.5.1	With the reactor subcritical less than 57 hours, verify one RHR loop is in operation and circulating reactor coolant at a flow rate of \geq [2800] 3000 gpm.	12' hours	3.9-9
	<u>OR</u> With the reactor subcritical for 57 hours or more, verify one RHR loop is in operation and circulating reactor coolant at a flow rate of \geq 1300 gpm.	12 hours	3.9-9

3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

NOTE
While this LCO is not met, entry into a MODE or other specified condition in the APPLICABILITY is not permitted.

3.9-12

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore required RHR loops to OPERABLE status.	Immediately
	<u>OR</u> A.2 Initiate action to establish \geq 23 ft of water above the top of reactor vessel flange.	Immediately
B. No RHR loop in operation.	B.1 Suspend operations involving a reduction in reactor coolant boron concentration. <u>AND</u>	Immediately (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Initiate action to restore one RHR loop to operation.	Immediately
	<u>AND</u> B.3 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.9.6.1	With the reactor subcritical less than 57 hours, verify one RHR loop is in operation and circulating reactor coolant at a flow rate of \geq [2800] 3000 gpm.	12 hours	<u>3.9-9</u>
	<u>OR</u> With the reactor subcritical for 57 hours or more, verify one RHR loop is in operation and circulating reactor coolant at a flow rate of \approx 1300 gpm.	12 hours	<u>B-PS</u> <u>3.9-9</u>
SR 3.9.6.2	Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days	

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LC0 3.9.7 Refueling cavity water level shall be maintained \geq 23 ft above the top of reactor vessel flange.

APPLICABILITY: ~~During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts,~~ During movement of irradiated fuel assemblies within containment.

3.9-10

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend CORE ALTERATIONS. <u>AND</u>	Immediately <u>3.9-10</u>
	A.2 <input checked="" type="checkbox"/> Suspend movement of irradiated fuel assemblies within containment. <u>AND</u>	Immediately
	A.3 Initiate action to restore refueling cavity water level to within limit.	Immediately <u>3.9-2</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.7.1 Verify refueling cavity water level is \geq 23 ft above the top of reactor vessel flange.	24 hours

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes- Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

**Methodology For Mark-up of NUREG-1431 Specifications
(Continued)**

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES CONTENTS

Mark-up:

<u>BASES</u>	<u>PAGE</u>
3.9.1	B 3.9-1
3.9.2	Not Used
3.9.3	B 3.9-4
3.9.4	B 3.9-7
3.9.5	B 3.9-11
3.9.6	B 3.9-14
3.9.7	B 3.9-17
Methodology	(1 Page)

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. The refueling boron concentration is sufficient to maintain shutdown margin (SDM) with the most adverse conditions of fuel assembly and control rod position allowed by plant procedures. The boron concentration that is maintained in Mode 6 is sufficient to maintain $k_{eff} \leq 0.95$ with the most reactive rod control assembly completely removed from its fuel assembly.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the principle system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with ~~refueling grade~~ borated water from the ~~liquid hold up tanks or the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.~~

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level") to provide forced circulation ~~cooling~~ in the RCS and assist in maintaining the boron concentrations ~~uniformity~~ in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

(Continued)

BASES

APPLICABLE SAFETY ANALYSIS

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5-3 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.2, "SHUTDOWN MARGIN (SDM) $T_c \leq 200^\circ F$." Boron dilution accidents are precluded in MODE 6 by isolating potential dilution flow paths. See LCO 3.9.2, "Unborated Water Source Isolation Valves." It is based upon a maximum dilution flow of 300 g.p.m. and prompt identification and operation preclude the event from proceeding to a boron dilution accident. Prompt identification is assured through audible count rate instrumentation, a high count rate alarm and a high source range flux level alarm.

The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement 10CFR50.36(c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM) $T_c \geq 200^\circ F$," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) $T_c \leq 200^\circ F$," LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical. A Note is added to the applicability to assure that MODE 6 cannot be entered unless boron concentration limits are met.

(Continued)

BASES

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE
REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1.10 CFR 50, Appendix A, GDC 26.

2. USAR Chapter ~~[15]~~ SAR, Chapter 15, Section 15.2.4

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range (source range drawer) covers six decades of neutron flux ($1E+6$ cps) (10 to $1E+6$ cps) with a $\pm 3\%$ instrument accuracy. The detectors also provide continuous visual indication in the control room and an audible alarm and count rate to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

The Gamma-Metrics neutron flux monitors (N-51 and N-52) are designed in accordance with Regulatory Guide 1.97. The wide range neutron flux monitors in this system provide indication of neutron flux from reactor shutdown to reactor full power level (source range through power range). The wide range monitors ($1E+8$ to $1E+2$ power) provide continuous visual indication in the control room to allow operators to monitor core flux. The narrow range monitors ($1E+1$ to $1E+5$ cps) provides indication of neutron flux to the hot shut down panel and control room by way of the plant process computer (PPC).

APPLICABLE SAFETY ANALYSIS Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. ~~The need for a safety analysis for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2, "Unborated Water Source Isolation Valves."~~ Prompt identification is required to assure sufficient time for operator action to preclude the event from proceeding to a Boron Dilution Accident. Prompt identification is assured through audible count rate instrumentation indication, a high count rate alarm and a high source range flux level alarm in the control room.

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement 10CFR50.36(c)(2)(iii).

LCO This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication and at least one of the two monitors must provide an audible alarm and count rate functions in the Control Room. Therefore, with no audible alarm and count rate functions from at least one monitor, both monitors are inoperable.

(Continued)

BASES (continued)

APPLICABILITY In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. The exception given in A.1 for the process of latching/unlatching control rods and friction testing of control rods is provided to allow completion of head installation prior to replacing a failed source range detector. RCGA latching and friction testing is conducted with the reactor vessel upper internals in place, thereby preventing the lowering of a temporary source range detector into the region of the core. This NOTE allows control rod movement with only one source range in place. Friction testing involves fully withdrawing and reinserting each rod in turn, which could change core reactivity by as much as one percent for the most reactive rod. The increase in count rate would be one to two counts per second. The core coupling in this configuration would allow one source range detector to detect significant reactivity changes associated with control rod movement (Ref 3). Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position or normal cooldown of a coolant volume for the purpose of system temperature control.

B.1

With no source range neutron flux monitor OPERABLE including no OPERABLE audible alarm and count rate functions, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor including no OPERABLE audible alarm and count rate functions is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

(Continued)

BASES (continued)

ACTIONS
(continued)

~~The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The Completion Time~~
Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

SURVEILLANCE
REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions. ~~For core reload, the first CHANNEL CHECK for each channel may be performed using the first fuel assembly as a source, prior to unlatching it in the core.~~

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. ~~The CHANNEL CALIBRATION also includes verification of the audible alarm and count rate functions on a simulated or actual boron dilution flux doubling signal.~~ The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
 2. FSAR, Section ~~15.4.6 [15.2.4]~~.
 3. ~~License Amendment 46/45, October, 1989~~
-
-

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed by automatic means. Since there is no any potential for containment pressurization yields very low levels, the 10CFR50 Appendix J leakage criteria and tests are not required. (Ref. 1)

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10CFR100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed for normal entry and exit.

(Continued)

BASES

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 42 36 48 inch purge penetration and a 42 36 48 inch exhaust penetration in which the flow path is limited to being open 200 hour or less per calendar year. The second subsystem, a minipurge pressure equalization system, includes an 8 18 inch purge penetration and an 8 18 provides a single 12 inch supply and exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and exhaust penetrations are secured in the closed position. The two valves in each of the two minipurge three valves in the 12 inch pressure equalization penetrations can be opened intermittently. Each of these system are qualified to but are closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchangers are necessary to conduct refueling operations. The normal 42 36 48 inch purge system is used for this purpose, and all four valves are closed by the ESFAS in accordance with LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation 3.3.6 Containment Purge and Exhaust Isolation Instrumentation".

The minipurge system remains operational in MODE 6, and all four valves are also closed by the ESFAS.

or

The minipurge pressure equalization system is not disassembled and used in MODE 6 for other outage functions. All four 8 inch valves are secured in the closed position.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. The fuel transfer tube is open but closure is provided by an equivalent isolation of a water loop seal. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements (Ref. 1).

APPLICABLE
SAFETY ANALYSIS

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 3 2, include consists

BASES

~~of dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).~~

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement ~~10CFR50.36(c)(2)(ii)~~.

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

~~LCO 3.9.4.c is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) Appropriate personnel are aware of the open status of the penetration flowpath during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment and 2) specified individuals are designated and readily available to isolate the flowpath in the event of a fuel handling accident.~~

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

(Continued)

BASES

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE
REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates by inspection or administrative means that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal. The SR specifies that containment penetrations that are open under administrative controls are not required to meet the SR during the time the penetrations are open.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.4.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.6, the Containment Purge and Exhaust Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is (continued) in accordance with the In service Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. ~~GPU Nuclear Safety Evaluation SE 0002000-001, Rev. 0, May 20, 1988.~~
~~Design Criteria Memorandum T-16, Containment Functions.~~
 2. FSAR, Section ~~[15.4.5]~~.
 3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.
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B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE
SAFETY ANALYSIS

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission product barrier.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement 10CFR50.36(c)(2)(iii) as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat

(Continued)

BASES

removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

- a. Removal of decay heat;
- b. Mixing of borated coolant to minimize the possibility of criticality; and
- c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as valve testing, core mapping, or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

The LCO is also modified by a second Note that allows the required RHR loop to be removed from service for up to 2 hours per 8 hour period to support surveillance leak rate testing of the RCS to RHR suction isolation valves, provided that no operations are permitted which might result in reduction of boron concentration. During this 2 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity and the RCS.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level $<$ 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Notes to the LCO.

(Continued)

BASES

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS because all of unborated water sources are isolated. The suspension of any operation involving a reduction in reactor coolant boron concentration will reduce the likelihood of stratification of the boron concentration developing within the RCS.~~

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a ~~an irradiated~~ fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level \geq 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE
REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate ~~of 3000 gpm~~ is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the

(Continued)

BASES

core prior to 57 hours of core subcriticality. The second part of this Surveillance serves the same function but with 57 hours or more of core subcriticality. The flow rate of 1300 gpm is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. Both of these flow rates are points of the same flow rate verses decay heat. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System (Ref. 2).

REFERENCES

1. FSAR, Section-~~[5.5.7]~~.
 2. LAR 88-01, dated 4/21/88, submitted by "RHR System Flow Rate Reduction," DCL 88-067.
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B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE
SAFETY ANALYSIS

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement ~~10CFR50.36(c)(2)(1)~~ as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

- a. Removal of decay heat;
- b. Mixing of borated coolant to minimize the possibility of criticality; and
- c. Indication of reactor coolant temperature. An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow

(Continued)

BASES

path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. One or both RHR pumps may be aligned to the RWST to support filling the refueling cavity or for performance of required testing (Ref. 2).

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level." A Note is added to the applicability to assure that MODE 6 operation with water level < 23 ft is not permitted unless two RHR loops are operable.

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS, because all of the unborated water sources are isolated.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop

(Continued)

BASES

requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE
REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate of more than 3000 gpm is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core prior to 57 hours subcritical. The second part of this Surveillance serves the same function but with 57 hours or more of core subcriticality and provides a reduced flow rate of 1300 gpm based upon a reduced decay heat load. Both of these flow rates are points of the same flow rate versus decay heat curve. The 1300 gpm limit also precludes exceeding the 1675 gpm upper flow limit to prevent vortexing and air entrainment of the RHR piping system. RHR pump vortexing (failure to meet pump suction requirements) during mid-loop operation may result in RHR pump failure and non-conservative RCS level indication. In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room (Ref 3).

SR 3.9.6.2

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. FSAR, Section ~~[5.5.7]~~
2. WOG Standard Technical Specification Change Traveler TSTF-21
3. [AR 88-01, dated 4/21/88, submitted by RHR System Flow Rate Reduction, DCL 88-067]

(Continued)

B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, or friction testing of individual control rods within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3 and acceptance in Reference 6.

APPLICABLE
SAFETY ANALYSIS

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained well within allowable limits (Refs. 4 and 5 and 6).

Refueling cavity water level satisfies Criterion 2 of the NRC Policy Statement 10CFR50.36(c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.

(Continued)

BASES

APPLICABILITY

LCO 3.9.7 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.15, "~~Spent Fuel Storage Pool~~ Water Level."

ACTIONS

~~A.1 and A.2~~

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving ~~CORE ALTERATIONS~~ or movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of ~~CORE ALTERATIONS~~ and fuel movement shall not preclude completion of movement of a component to a safe position.

~~A.3~~

~~In addition to immediately suspending CORE ALTERATIONS or movement of irradiated fuel, action to restore refueling cavity water level must be initiated immediately.~~

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide 1.25, March 23, 1972.
 2. FSAR, Section ~~[15.7.4.5]~~.
 3. NUREG-0800, Section 15.7.4.
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(Continued)

BASES

4. 10 CFR 100.10.
 5. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J.,
WCAP-828, Radiological Consequences of a Fuel Handling
Accident, December 1971.
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Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(2 Pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.9

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>JUSTIFICATION</u>
3.9-1	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 6B).
3.9-2	ITS 3.9.7, Action A.3 to restore water level is unnecessary once fuel movement has ceased. The specification would not allow additional movement of irradiated fuel without restoring water level. Additionally, Required Action A.3 is unnecessary since completion of ITS Required Action A.1 results in exiting the Mode of Applicability. This is consistent with TSTF-20.
3.9-3	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.9-4	Consistent with the CTS, LCO 3.9.3 Required Action B.2 is revised in accordance with traveler TSTF-96. The initial performance of ITS SR 3.9.1.1 within 4 hours of entry into Condition B of LCO 3.9.3 is proposed for deletion. The accelerated performance of this SR is not warranted based on routine performance of this SR (every 72 hours), and knowledge of stable conditions prior to the loss of the source range monitor. Secondly, RCS dilution events are recognizable through other means such as controlled increases in pool water level.
3.9-5	Required Action A is expanded to incorporate current licensing bases allowing latching of control rod drive shafts and friction testing of individual control rod with one source range flux monitor inoperable.
3.9-6	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.9-7	In accordance with CTS, LCO 3.9.4 status item c.2 and associated Bases, would be modified to state "capable of being closed by an OPERABLE [Containment Purge and Exhaust Isolation Valve]" (i.e. the word "valve" would replace the word "system").
3.9-8	The Note of ITS 3.9.5 is expanded to incorporate current licensing bases allowing the required RHR pump to be removed from service for 2 hours per 8 hours for testing of leak testing of the RHR suction isolation valves.
3.9-9	The Surveillances of ITS 3.9.5 and 3.9.6 are modified to incorporate current licensing bases of two RHR flows dependent upon the number of hours the reactor has been subcritical.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.9

CHANGE NUMBER

JUSTIFICATION

- 3.9-10 Consistent with the CTS, ITS 3.9.7 Applicability is being revised to delete "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts." The Applicability is being revised because this requirement is duplicative of a relocated technical specification requirement for reactor vessel water level during movement of control rods (relocated technical specification 3.9.10.2). The relocated specification places a lower limit on the amount of water above the top of the fuel assemblies in the reactor vessel during movement of control rods. The Bases (CTS 3/4.9.10) states that this ensures the water removes 99% of the assured 10% iodine gap activity released from the rupture of an irradiated fuel assembly in the event of a fuel handling accident (FHA). However, the movement of control rods is not associated with the initial conditions of a FHA, and the Bases do not address any concern regarding inadvertent criticality which could lead to a breach of the fuel rod cladding.
- 3.9-11 In accordance with a proposed traveler WOG-76, LCO is modified to permit a penetration flow path that provides direct access from the containment to the outside atmosphere to be unisolated under administrative controls. The allowance to have containment flow paths with direct access from the containment atmosphere to the outside atmosphere unisolated under administrative controls is based on confirmatory dose calculations of a fuel handling accident which indicate acceptable radiological consequences and to implement administrative controls that ensure that the flow penetrations will be promptly closed following a fuel handling accident, to provide a defense-in-depth approach to meet acceptable dose consequences. The administrative control requirements are defined in the Bases.
- 3.9-12 A note is added to the Applicability of LCO 3.9.6 indicating that entry into a MODE or other specified condition in the Applicability is not permitted while the LCO is not met. The addition of this note is based on the performance of a plant-specific LCO 3.0.4 matrix which identified where the requirements of 3.0.4 are still applicable in MODES 5 and 6 and in MODES 1, 2, 3, and 4 when the MODE is descending (i.e., from MODE 1 to MODE 2, etc.). This matrix was specified in the NUREG-1431 reviewer's note in LCO 3.0.4.
- 3.9-13 In accordance with DCPD CTS, LCO 3.9.2 would not be used. This new requirement is not applicable to DCPD which has a licensed dilution accident. The current licensing bases in accordance with NUREG 0800, Section 15.4.6 provides adequate assurance that a dilution event will be recognized and arrested in a timely fashion.
- 3.9-14 A Note is added to the Applicability of LCO 3.9.1 indicating that entry into MODE 6 from MODE 5 is not permitted while the LCO is not met. The addition of this Note is based on the performance of a plant-specific LCO 3.0.4 matrix which identified where the requirements of 3.0.4 are still applicable in MODES 5 and 6 and in MODES 1, 2, 3, and 4 when the MODE is descending (i.e., from MODE 1 to MODE 2, etc.). This matrix was specified in the NUREG-1431 reviewer's note in LCO 3.0.4.

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(2 Pages)

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.9-1	Revises ITS 3.9.4, LCO b. to permit the one door of the emergency airlock to remain open and both doors of the personnel air lock to remain open with one capable of closure during core alterations or movement of irradiated fuel.	No, Not permitted by CTS.	Yes	Yes	Yes
3.9-2	ITS 3.9.7, ACTION A.3 to restore water level is unnecessary once fuel movement has ceased. The specification would not allow additional movement of irradiated fuel without restoring water level. This is consistent with TSTF-20.	Yes	Yes	Yes	Yes
3.9-3	An additional ACTION to suspend positive reactivity changes, if a boron dilution source is not isolated, has been added to match the CTS.	No, not in CTS	Yes	No, not in CTS	No, not in CTS
3.9-4	Revise LCO 3.9.3 Required Action B.2 in accordance with traveler TSTF-96. The initial performance of ITS SR 3.9.1.1 within 4 hours of entry into Condition B of LCO 3.9.3 is proposed for deletion.	Yes	Yes	Yes	Yes
3.9-5	Required Action A.1 is expanded to incorporate CTS allowing latching of control rod drive shafts and friction testing of individual control rod with one source range flux monitor inoperable.	Yes, See LAR 89-09 dated 10/30/89 and DCL 89-213	No, not in CTS	No, not in CTS	No, not in CTS
3.9-6	A Note is added to LCO 3.9.4 stating that an emergency escape hatch temporary closure device is an acceptable replacement for the airlock door. This change is consistent with CTS.	No, not in CTS.	No, not in CTS	Yes, amendment 74	No, not in CTS
3.9-7	In accordance with CTS, LCO 3.9.4 status item c.2 and associated Bases, would be modified to state "capable of being closed by an OPERABLE [Containment Purge and Exhaust Isolation Valve]" (i.e. the word "valve" would replace the word "system").	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.9

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.9-8	The Note of ITS 3.9.5 is expanded to incorporate CTS allowing the required RHR pump to be removed from service for less than or equal to 2 hours per 8 hours for leak testing of the RHR suction isolation valves.	Yes	No, not in CTS	No, not in CTS	No, not in CTS
3.9-9	The Surveillances of ITS 3.9.5 and 3.9.6 are modified to incorporate CTS of two RHR flow rates dependent upon the number of hours the reactor has been subcritical.	Yes, See LAR 88-01 dated 4/21/88 and DCL 88-067	No, not in CTS	No, not in CTS	No, not in CTS
3.9-10	ITS 3.9.7 Applicability is being revised to delete "During CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts." The Applicability is being revised because this requirement is duplicative of a relocated technical specification requirement for reactor vessel water level during movement of control rods.	Yes	Yes	Yes, relocated per Amendment 89	Yes, relocated per Amendment 103
3.9-11	In accordance with traveler WOG-76, LCO is modified to permit a penetration flow path that provides direct access from the containment to the outside atmosphere to be unisolated under administrative controls.	Yes	Yes	Yes	Yes
3.9-12	A Note is added to the Applicability of LCO 3.9.6 indicating that entry into a MODE or other specified condition in the Applicability is not permitted while the LCO is not met.	Yes	Yes	Yes	Yes
3.9-13	In accordance with DCPD CTS, LCO 3.9.2 would not be used	Yes	No	No	No
3.9-14	A Note is added to the Applicability of LCO 3.9.1 indicating that entry into MODE 6 from MODE 5 is not permitted while the LCO is not met.	Yes	Yes	Yes	Yes

JLS Conversion to Improved Technical Specifications

Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 3/4.10 - Special Test Exceptions

CTS 3/4.11 - Radioactive Effluents



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.10

CONTENTS

- ENCLOSURE 1 - CROSS-REFERENCE TABLES
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- ENCLOSURE 4 - NO SIGNIFICANT HAZARDS CONSIDERATIONS
- ENCLOSURE 5A - MARK-UP OF NUREG-1431 SPECIFICATIONS
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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(1 Pages)
CONVERSION TABLE SORTED BY IMPROVED TS	(1 Pages)
METHODOLOGY	(3 Pages)

CROSS-REFERENCE TABLE FOR 3/4.10
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.10.1			1-01-M				
3.10.2			2-01-M				
3.10.3	LCO			3.1.8			3.1-9 3.1-15
3.10.3	LCO	a		3.1.8	LCO	c	3.1-13
3.10.3	LCO	b	3-01-M 3-02-A	3.1.8	LCO	b	3.1-1
3.10.3	LCO	c		3.1.8	LCO	a	3.1-20
3.10.3	APP		3-04-A	3.1.8	APP		3.1-13
3.10.3	Action	a		3.1.8	Action	b	
3.10.3	Action	b		3.1.8	Action	c	3.1-20
3.10.3	Action	b		3.1.8	Action	d	
3.10.3	Action	c "New"	3-01-M	3.1.8	Action	a	
4.10.3.1	SR			3.1.8.3	SR		3.1-13
4.10.3.2	SR			3.1.8.1	SR		
4.10.3.3	SR			3.1.8.2	SR		3.1-20
"New"	SR		3-01-M	3.1.8.4	SR		3.1-1
3.10.4			5-01-R				

CROSS-REFERENCE TABLE FOR 3/4.10
Sorted by Improved TS

Improved TS				Current TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.1.8			3.1-9 3.1-15	3.10.3	LCO		
3.1.8	LCO	a	3.1-20	3.10.3	LCO	c	
3.1.8	LCO	b	3.1-1	3.10.3	LCO	b	3-01-M 3-02-A
3.1.8	LCO	c	3.1-13	3.10.3	LCO	a	
3.1.8	APP		3.1-13	3.10.3	APP		3-04-A
3.1.8	Action	a		3.10.3	Action	c "New"	3-01-M
3.1.8	Action	b		3.10.3	Action	a	
3.1.8	Action	c	3.1-20	3.10.3	Action	b	
3.1.8	Action	d		3.10.3	Action	b	
3.1.8.1	SR			4.10.3.2	SR		
3.1.8.2	SR		3.1-20	4.10.3.3	SR		
3.1.8.3	SR		3.1-13	4.10.3.1	SR		
3.1.8.4	SR		3.1-1	"New"	SR		3-01-M

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

**Methodology for Cross-Reference Tables
(Continued)**

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).
Note:	The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
3.10.1	3/4 10-1
3.10.2,	3/4 10-2
3.10.3	3/4 10-3
3.10.4	3/4 10-4
Methodology	(2 Pages)

~~3/4.10.1 SHUTDOWN MARGIN~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).~~

~~APPLICABILITY: MODE 2.~~

~~ACTION:~~

- ~~a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for the trip insertion immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.~~
- ~~b. With all full length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.1.1 The position of each full length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.~~

~~4.10.1.2 Each full length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.~~

SPECIAL TEST EXCEPTIONS

02-01-M

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

~~3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:~~

- ~~a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and~~
- ~~b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2.~~

~~APPLICABILITY: MODE 1.~~

ACTION:

~~With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:~~

- ~~a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or~~
- ~~b. Be in HOT STANDBY within 6 hours.~~

SURVEILLANCE REQUIREMENTS

~~4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.~~

~~4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:~~

- ~~a. Specifications 4.2.2.2 and 4.2.2.3, and~~
- ~~b. Specification 4.2.3.2.~~

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

- a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER.
- b. ~~The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and SHUTDOWN MARGIN is within the limits provided in the COLR and~~ 03-02-A
03-01-M
- c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 531°F.

APPLICABILITY: ~~MODE 2 during PHYSICS TESTS.~~ 03-04-A

ACTION:

- a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.
- b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 531°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

~~With SHUTDOWN MARGIN not within its limits, within 15 minutes initiate boration to restore SHUTDOWN MARGIN to within its limits and, within one hour, suspend PHYSICS TESTS exception.~~ 03-01-M

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to a CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 531°F at least once per 30 minutes during PHYSICS TESTS.

~~(NEW) Verify SHUTDOWN MARGIN to be within limits provided on the COLR once per 24 hours.~~ 03-01-M

~~3/4 10-4 POSITION INDICATION SYSTEM SHUTDOWN~~

LIMITING CONDITION FOR OPERATION

~~3.10.4 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full length shutdown and control rod drop time measurements provided only one shutdown or control bank is withdrawn from the fully inserted position at a time.~~

~~APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements and during surveillance of digital rod position indicators for OPERABILITY.~~

ACTION:-

~~With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.~~

SURVEILLANCE REQUIREMENTS

~~4.10.4 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:-~~

- ~~a. Within 12 steps when the rods are stationary, and~~
- ~~b. Within 24 steps during rod motion.~~

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. **Deletions** - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. **Additions** - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. **Modifications** - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. **Administrative** - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions** - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions** - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications** - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative** - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

Methodology For Mark-Up of Current TS
(Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers (1 Page)

Description of Changes (2 Pages)

TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS

SECTION 3.4.10

Technical Specification Title	CHG. NO.	Callaway	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Shutdown Margin	01	None	None	3.10.1	3.10.1
Group Height, Insertion, and Power Distribution Limits	02	3.10.2	3.10.2	3.10.2	3.10.2
Physics Tests	03	3.10.3	3.10.3	3.10.3	3.10.3
Reactor Coolant Loops	04	3.10.4	3.10.4	3.10.4	None
Position Indication System - Shutdown	05	None	None	3.10.5	3.10.4

DESCRIPTION OF CHANGES TO TS SECTION 3/4.10

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	M	Limiting Condition for Operation (LCO) 3.10.1 would be deleted consistent with Traveler TSTF-12, Rev. 1. The SHUTDOWN MARGIN (SDM) special test exception (STE) requires rod worth measurement in the N-1 condition. Other rod worth measurements techniques will maintain the SDM during the measurement process and provide the necessary physics data. The N-1 measurement technique is no longer used.
02-01	M	The MODE 1 STE LCO 3.10.2 would be deleted in accordance with Traveler TSTF-12, Rev. 1. TSTF-12, Rev. 1 would delete improved Technical Specification (ITS) LCO 3.1.9, which contains the STEs for MODE 1, because the tests implemented by ITS LCO 3.1.9 were contained in some initial plant startup test programs and are not performed during post-refueling startup testing. Post-refueling tests may be performed in accordance with other LCOs. This change is acceptable because it imposes more stringent requirements (i.e., eliminating exceptions). The elimination of the STE has no adverse impact on the health and safety of the public.
03-01	M	Consistent with NUREG-1431, the SDM parameter is added to the list of preconditions required prior to invoking this STE. The actual value of the shutdown margin is located in the Core Operating Limits Report (COLR). Associated ACTION statements and surveillances are also added. This change is acceptable because it imposes more stringent TS requirements that are both appropriate and consistent with NUREG-1431, Rev. 1.
03-02	A	The requirement to ensure that the reactor trip setpoints are OPERABLE already exists in the Reactor Trip System LCO, Table 3.3.1-1 functional items 2 and 4 of the ITS which are applicable in the same MODE. This change is acceptable because there is no need to reference another LCO if that LCO is applicable in the same MODE.
03-03		Not Used.
03-04	A	The APPLICABILITY statement would be changed to be more consistent with operation for testing purposes. The proposed change is consistent with NUREG-1431, Rev. 1, and does not result in any changes to technical requirements.

DESCRIPTION OF CHANGES TO TS SECTION 3/4.10
(Continued)

CHANGE

NUMBER

NSHC

DESCRIPTION

04-01

M

Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).

05-01

R

STE LCO 3.10.4, "Position Indication System - Shutdown," would be relocated based on relocation of LCO 3.1.3.3, "Position Indicating Systems - Shutdown." NRC application of TS Criteria concluded that the STE LCOs could be included with corresponding LCOs remaining in TS and that LCO 3.10.4 could be relocated with LCO 3.1.3.3.

5-02

M

Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(2 pages)

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.10

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 M	LCO 3.10.1 would be deleted consistent with Traveler TSTF-12. The SDM STE requires rod worth measurement in the N-1 condition. Other rod worth measurements techniques will maintain the SDM during the measurement process and provide the necessary physics data. The N-1 measurement technique is no longer used.	Yes, see Attachment 21, page 29	Yes	No, Amendment 89 previously deleted this STE.	No, Amendment 103 previously deleted this STE.
02-01 M	The MODE 1 STE LCO 3.10.2 would be deleted in accordance with Traveler TSTF-12. TSTF-12 would delete ITS LCO 3.1.9, which contains the STEs for MODE 1 because the tests implemented by ITS LCO 3.1.9 were contained in some initial plant startup test programs and are not performed during post-refueling startup testing. Post-refueling tests may be performed in accordance with other LCOs.	Yes	Yes	Yes	Yes
03-01 M	Consistent with NUREG-1431, Rev. 1, the SDM parameter is added to the list of preconditions required prior to invoking this STE. The actual value of the SDM is located in the COLR. Associated ACTION statements and surveillances are also added.	Yes	Yes	Yes	Yes
03-02 A	The requirement to ensure that the reactor trip setpoints are operable already exists in the reactor trip system LCO, Table 3.3.1-1, functional items 3 and 4 of the ITS.	Yes	Yes	Yes	Yes
03-03	Not Used.	NA	NA	NA	NA

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
04-01 M	STE LCO [3.10.4], "Reactor Coolant Loops," would be deleted. This specification allows the suspending of requirements of one or more LCOs (depending on plant-specific current TS) under certain Conditions. Elimination of the STEs is justified either because their elimination would be consistent with NUREG-1431, Rev 1, or because the applicable tests are performed only during initial plant startup and are no longer needed. Therefore, because the exceptions applicable to each LCO addressed by LCO [3.10.4] are no longer required, LCO [3.10.4] may be eliminated.	No, not in CTS	Yes	Yes	Yes
05-01 R	STE LCO 3.10.4, "Position Indication System Shutdown," would be relocated based on relocation of LCO 3.1.3.3, "Position Indicating Systems - Shutdown." NRC application of TS criteria concluded that the STE LCOs could be included with corresponding LCOs remaining in TS and that LCO 3.10.4 could be relocated with LCO 3.1.3.3.	Yes, see Attachment 21, page 31	No, see CN 5-02-M	No, relocated per Amendment 89.	No, relocated per Amendment 103.
05-02 M	STE LCO 3.10.5, "Position Indication System Shutdown," would be deleted based on relocation of LCO 3.1.3.3, "Position Indication System - Shutdown." NRC application of TS criteria concluded that the STE LCOs could be included with corresponding LCOs remaining in TS, and thus LCO 3.10.5 is no longer necessary.	No, see CN 5-01-R	Yes	No, relocated per Amendment 89	No, relocated per Amendment 103

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

	<u>PAGE</u>
I. Organization	2
II. Description of NSHC Evaluations	3
III. Generic NSHCs	
"A" - Administrative Changes	5
"R" - Relocated Technical Specifications	7
"LG" - Less Restrictive (moving information out of the TS)	10
"M" - More Restrictive	12
IV. Specific NSHCs - "LS"	None

I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3...." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR

ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers (1 Page)

NUREG-1431 Specifications that are not applicable (1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
3.1.8	3.1-23
3.1.11	N/A
Methodology	(2 Pages)

Industry Travelers Applicable to Section 3.10

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated	3.1-1	Approved by NRC
TSTF-12, Rev. 1	Incorporated	3.1-15	Approved by NRC
TSTF-14, Rev. 3	Incorporated	3.1-13	As approved by the NRC
TSTF-108	Not Incorporated	N/A	NRC Rejected
TSTF-136	incorporated	3.1-15	

NUREG-1431 SPECIFICATIONS THAT ARE NOT APPLICABLE

Specification #	Specification Title	Comments
3.1.9	PHYSICS TEST Exceptions, Mode 1	Industry traveler TSTF-12 deletes LCO 3.1.9 which contains the STEs for Mode 1 because the tests implemented by LCO 3.1.9 were contained in some initial plant startup test programs and are not performed during post-refueling startup testing. Post-refueling tests may be performed in accordance with other LCOs. This change renumbers ITS 3.1.10 to 3.1.9.
3.1.11	SHUTDOWN MARGIN Test Exceptions	This TS was proposed for relocation in LAR 96-01 (see change number 01-01-M in Enclosure 3B).
3.4.19	RCS Loops - Test Exceptions	DCPP TS do not contain this LCO.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.408 PHYSICS TESTS Exceptions—MODE 2

LCO 3.1.408 During the performance of PHYSICS TESTS, the requirements of

- LCO 3.1.43, "Moderator Temperature Coefficient (MTC)";
- LCO 3.1.54, "Rod Group Alignment Limits";
- LCO 3.1.65, "Shutdown Bank Insertion Limits";
- LCO 3.1.76, "Control Bank Insertion Limits"; and
- LCO 3.4.2, "RCS Minimum Temperature for Criticality"

3.1-9

may be suspended, provided:

- a. RCS lowest operating loop average temperature is \geq [531] °F; and
- b. SDM is \geq [1.6] % $\Delta k/k$, within the limits provided in the COLR; and
- c. THERMAL POWER is \leq 5% RTP

3.1-20

B

3.1-1

3.1-13

APPLICABILITY: During PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit. AND A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS lowest operating loop average temperature not within limit.	C.1 Restore RCS lowest operating loop average temperature to within limit.	15 minutes <u>3.1-20</u>
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.108.1 Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per [SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1].	Within 12 hours prior to initiation of PHYSICS TESTS
SR 3.1.108.2 Verify the RCS lowest operating loop average temperature is ≥ 531 °F.	30 minutes <u>3.1-20</u> <u>B</u>
SR 3.1.8.3 Verify THERMAL POWER is $\geq 5\%$ RTP	1 hour <u>3.1-13</u>
SR 3.1.108.34 Verify SDM is $\rightarrow 1.6\%$ $\Delta k/k$. within limits provided in the COLR	24 hours <u>3.1-1</u>

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions -** The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions -** The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications -** The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes-** Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts -** The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

Methodology For Mark-up of NUREG-1431 Specifications
(Continued)

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

MARK-UP OF NUREG-1431 BASES CONTENTS

Mark-up:

<u>BASES</u>	<u>PAGE</u>
3.1.8	B3.1-60
3.1.11	N/A
Methodology	(1 Page)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.108 PHYSICS TESTS Exceptions-MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

- a. Ensure that the facility has been adequately designed;
- b. Validate the analytical models used in the design and analysis;
- c. Verify the assumptions used to predict unit response;
- d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and
- e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed.

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

BASES

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below: typically include:

- a. Critical Boron Concentration - Control Rods Withdrawn;
- b. Critical Boron Concentration - Control Rods Inserted;
- c. Control Rod Worth; and

BASES

- d. Isothermal Temperature Coefficient (ITC); and
- e. ~~Neutron Flux Symmetry.~~

~~The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.~~

- a. ~~The Critical Boron Concentration Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical ($k_{eff} = 1.0$), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.~~
- b. ~~The Critical Boron Concentration Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron~~

BASES

BACKGROUND
 (continued)

- ~~concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.5, "Rod Group Alignment Limits"; LCO 3.1.6, "Shutdown Bank Insertion Limit"; or LCO 3.1.7, "Control Bank Insertion Limits."~~
- e. ~~The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.5, LCO 3.1.6, or LCO 3.1.7.~~
- d. ~~The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of~~

BASES

BACKGROUND
(continued)

~~performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."~~

- ~~e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at $\leq 30\%$ RTP (Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.5, LCO 3.1.6, or LCO 3.1.7. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at $\leq 30\%$ RTP.~~

APPLICABLE
SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

BASES

APPLICABLE
SAFETY ANALYSES
(Continued)

The FSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables [14.1-1 and 14.1-2] summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS 19.6.1-1985 (Ref. 4). Reload fuel cycle PHYSICS TESTS are performed in accordance with Technical Specification requirements, fuel vendor guidelines, and established industry practices. Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design

BASES

criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to $\leq 5\%$ RTP, the reactor coolant temperature is kept $\geq 531^\circ\text{F}$, and SDM is $\geq [1.6]\% \text{ Ak/k}$ within the limits provided in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. ~~Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses.~~ Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement 10 CFR 50.36(c)(2)(iii).

Reference 6 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond LCO specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.4 ~~3~~, LCO 3.1.5 ~~4~~, LCO 3.1.6 ~~5~~, LCO 3.1.7 ~~6~~, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

- a. RCS lowest ~~operating~~ loop average temperature is $\geq [531]^\circ\text{F}$; ~~and~~
- b. SDM is $\geq [1.6]\% \text{ Ak/k}$ within the limits provided in the COLR; ~~and~~

~~c. THERMAL POWER IS $\leq 5\%$ RTP~~

APPLICABILITY

This LCO is applicable ~~in MODE 2~~ when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP. ~~Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, "PHYSICS TESTS Exceptions MODE 1."~~

BASES

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest ~~operating loop's~~ T_{avg} is < 531°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with ~~an operating loop's~~ temperature below 531°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.108.1

The ~~required~~ power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is

(Continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

performed on each OPERABLE power range and intermediate range channels within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.108.2

Verification that the RCS lowest operating loop T_{avg} is $\geq 531^{\circ}F$ will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.108.3

Verification that the THERMAL POWER is $\leq 5\%$ RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of ~~30 minutes~~ 1 hour during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.108.4

Verification that the SDM is within limits specified in the GCLR ensures that, for the specific RCCA and RCS temperature manipulations performed during PHYSICS TESTS, the plant is not operating in a condition that could invalidate the safety analysis assumptions.

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

- ~~a. RCS boron concentration;~~
- ~~b. Control bank position;~~
- ~~c. RCS average temperature;~~
- ~~d. Fuel burnup based on gross thermal energy generation;~~
- ~~e. Xenon concentration;~~
- ~~f. Samarium concentration; and~~
- ~~g. Isothermal temperature coefficient (ITC).~~

BASES

~~Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.~~

~~The SDM for physics testing during tests where traditional SDM monitoring techniques are not adequate, is determined for the most restrictive test based on design calculations. Plant conditions are monitored during these tests to verify adequate SDM.~~

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
 2. 10 CFR 50.59.
 3. Regulatory Guide 1.68, Revision 2, August, 1978.
 4. ~~ANSI/ANS 19.6.1 1985, December 13, 1985. Not used~~
 5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.
 6. WCAP-11618, including Addendum 1, April 1989.
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Methodology For Mark-up of NUREG-1431 Bases

Enclosure 5B contains an electronic (or hand written) mark-up of the Bases portion of NUREG 1431, Rev. 1. The Bases is descriptive in nature but provides significant clarification and, in some cases, technical information which supports the specifications. The version in the NUREG is generic while the improved TS version has been made plant specific.

To the extent possible, the words of NUREG 1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading, they have been corrected. In addition, descriptions have been added to cover plant specific portions of the specifications.

The changes are processed as follows:

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1 Bases.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1 Bases.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 Bases but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS or other design basis document into a bracketed portion of NUREG-1431, Rev. 1 Bases.

The methodology of identifying the changes is :

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are not identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 Bases is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is not identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 Bases is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the Bases in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is not identified by an item number or a change code in the adjacent right margin.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct, the "generic" information is "struck-out" and the correct material is inserted using the "red-line" feature. If the "generic" is correct, the information is "red-lined." The brackets are also deleted. An identification number to cross-reference to an explanation or justification is not provided.

Note: All brackets are deleted from the mark-up of NUREG-1431, Rev. 1 Bases as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred

In summary, "red-line" (or hand written/insert pages) is used to annotate new material, "strike-out" (or crossed out by hand) is used to annotate deleted material. Neither identification numbers nor change codes are used to identify changes in the Bases.

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(1 Page)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.10

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE NUMBER

JUSTIFICATION

- 3.1-1 In accordance with TSTF-9, Rev. 1 this change would relocate the specified limit for SDM from the ISTS to the COLR. This change occurs in several specifications including the specification for SDM and those specifications with ACTIONS that require verifying SDM within limits.
- 3.1-9 This change would eliminate ISTS 3.1.2 because the SDM requirements for MODE 5 have been incorporated into Specification 3.1.1 in accordance with traveler TSTF-136. Traveler TSTF-9, Rev. 1, relocated values for SDM to the COLR which removed the only difference between ITS LCO 3.1.1 and ITS LCO 3.1.2. Differences above and below 200 °F will be addressed in the COLR. Subsequent sections have been renumbered.
- 3.1-13 This change adds an LCO requirement and SR to MODE 2 Physics Tests Exceptions 3.1.8 to verify that rated RATED THERMAL POWER (RTP) is less than or equal to 5 percent RTP. The LCO requirement and SR were added to verify that RTP is within the defined power level for MODE 2 during the performance of physics tests, since there is an ACTION that addresses RTP not within limit yet there was no corresponding LCO or SR. The surveillance frequency of one hour is retained from the CTS. This change is based on Traveler TSTF-14, Rev. 3.
- 3.1-14 Not Used.
- 3.1-15 Consistent with TSTF-12, Rev. 1, LCOs 3.1.9 and 3.1.11 are deleted. The physics tests contained in LCO 3.1.9 were only contained in some plant initial plant startup testing programs. The physics test can be deleted since these physics tests are never performed during post-refueling outages. The physics test that LCO 3.1.11 required was the rod worth measurement in the N-1 condition. The use of other rod worth measurement techniques will maintain the SDM during the entire measurement process and still provide the necessary physics data verification. Since the N-1 measurement technique is no longer used, the SDM test exception can be deleted. This change and Traveler TSTF-136 renumbers ITS 3.1.10 to 3.1.8.
- 3.1-20 Consistent with CTS 3/4.10.3, "Physics Tests," ITS LCO 3.1.8 and its Condition C and SR 3.1.8.2 are modified to refer to "operating" RCS loops. Adopting the CTS is acceptable since valid T_{avg} measurements are not obtainable for a non operating loop.

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(1 Page)

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 3.10

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.1-1	In accordance with Traveler TSTF-9, Rev. 1, this change would relocate the specified limits for SDM from several TS to the COLR.	Yes	Yes	Yes	Yes
3.1-9	This change would eliminate ITS 3.1.2 because the SDM requirements for MODE 5 have been incorporated into Specification 3.1.1 in accordance with Traveler TSTF-136.	Yes	Yes	Yes	Yes
3.1-13	In accordance with Traveler TSTS-14, Rev. 3, the LCO and SR are modified to verify that thermal power \leq 5%. This provides an LCO requirement to correspond to Condition B which requires RTP to be within limit.	Yes	Yes	Yes	Yes
3.1-14	Not Used.	NA	NA	NA	NA
3.1-15	In accordance with Traveler TSTF-12, Rev. 1, this change would delete ISTS LCO 3.1.9 and 3.1.11. This change and TSTF-136 rennumbers ISTS 3.1.10 to ISTS 3.1.8.	Yes	Yes	Yes	Yes
3.1-20	Consistent with CTS 3/4.10.3, "Physics Tests," ITS LCO 3.1.8 and its Condition C and SR 3.1.8.2 are modified to refer to "operating" RCS loops.	Yes	Yes	Yes	Yes

IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 3/4.11

CONTENTS

- ENCLOSURE 1 - CROSS-REFERENCE TABLES
- ENCLOSURE 2 - MARK-UP OF CURRENT TS
- ENCLOSURE 3A - DESCRIPTION OF CHANGES TO CURRENT TS
- ENCLOSURE 3B - CONVERSION COMPARISON TABLE - CURRENT TS
- ENCLOSURE 4 - NO SIGNIFICANT HAZARDS CONSIDERATIONS
- ENCLOSURE 5A - MARK-UP OF NUREG-1431 SPECIFICATIONS
- ENCLOSURE 5B - MARK-UP OF NUREG-1431 BASES
- ENCLOSURE 6A - DIFFERENCES FROM NUREG-1431
- ENCLOSURE 6B - CONVERSION COMPARISON TABLE - NUREG 1431

ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(1 Page)
CONVERSION TABLE SORTED BY IMPROVED TS	(N/A)
METHODOLOGY	(3 Pages)

CROSS-REFERENCE TABLE FOR 3/4.11
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
3.11.1.4	LCO		1-01-LG	NA			
3.11.1.4	APP		1-01-LG	NA			
3.11.1.4	Action	a	1-01-LG	NA			
3.11.1.4	Action	b	1-01-LG	NA			
4.11.1.4	SR		1-01-LG	NA			
3.11.2.5	LCO		1-02-LG	NA			
3.11.2.5	APP		1-02-LG	NA			
3.11.2.5	Action	a	1-02-LG	NA			
3.11.2.5	Action	b	1-02-LG	NA			
3.11.2.5	Action	c	1-02-LG	NA			
4.11.2.5	SR		1-02-LG	NA			
3.11.2.6	LCO		1-03-LG	NA			
3.11.2.6	APP		1-03-LG	NA			
3.11.2.6	Action	a	1-03-LG	NA			
3.11.2.6	Action	b	1-03-LG	NA			
4.11.2.6	SR		1-03-LG	NA			

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

**Methodology for Cross-Reference Tables
(Continued)**

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
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Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).
Note:	The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
3/4.11.1.4	3.11-1
3/4.11.2.5.	3.11-2
3/4.11.2.6	3.11-3
Methodology	(2 Pages)

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 - 3.11.1.3 Deleted

3.11.1.4 The quantity of radioactive material contained in any temporary outdoor tanks shall be ~~limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases in accordance with Explosive Gas and Storage Tank Radioactivity Monitoring Program. The limits for the liquid radwaste quantities shall be maintained such that 10 CFR Part 20 limits are met.~~

01-01-LG

APPLICABILITY: ~~At all times.~~

ACTION:

- a. ~~With the quantity of radioactive material in any of the temporary outdoor tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.6.~~
- b. ~~The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

4.11.1.1 - 4.11.1.3 Deleted

4.11.1.4 The quantity of radioactive material contained in each of the temporary outdoor tanks shall be ~~determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank, in accordance with Explosive Gas and Storage Tank Radioactivity Monitoring Program to ensure that the quantity of radioactivity contained in temporary outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA, in the event of an uncontrolled release of the tanks' contents.~~

01-01-LG

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.1 - 3.11.2.4 Deleted

01-02-LG

3.11.2.5 The concentration of oxygen in the GASEOUS RADWASTE SYSTEM shall be ~~limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.~~ in accordance with Explosive Gas and Storage Tank Radioactivity Monitoring Program

APPLICABILITY: ~~At all times.~~

ACTION:

- a. ~~With the concentration of oxygen in the GASEOUS RADWASTE SYSTEM greater than 2% by volume but less than or equal 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.~~
- b. ~~With the concentration of oxygen in the GASEOUS RADWASTE SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.~~
- c. ~~The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

4.11.2.1 - 4.11.2.4 Deleted

4.11.2.5 The concentration of hydrogen* and oxygen in the GASEOUS RADWASTE SYSTEM shall be determined to be within the above limits by monitoring the waste gases in the GASEOUS RADWASTE SYSTEM with the hydrogen and continuous oxygen monitors required OPERABLE by Table 3.3.13 of Specification 3.3.3.10. in accordance with Explosive Gas and Storage Tank Radioactivity Monitoring Program to ensure limits are maintained which are appropriate to the system's design criteria

01-02-LG

~~*If monitoring of the waste gases for hydrogen is not performed, the hydrogen concentration shall be assumed to be greater than 4% by volume.~~

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 10^5 curies noble gases (considered as Xe 133 equivalent) in accordance with Explosive Gas and Storage Tank Radioactivity Monitoring Program. The gaseous radioactivity quantities shall be determined following the methodology in Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure" 01-03-LG

APPLICABILITY: At all times.

ACTION:-

- a. ~~With the quantity of radioactive material in any gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.6.~~
- b. ~~The provisions of Specification 3.0.3 are not applicable.~~

3.11.3 Deleted

3.11.4 Deleted

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once 7 days when radioactive materials are being added to the tank and at least once per 24 hours during primary coolant system degassing operations. in accordance with Explosive Gas and Storage Tank Radioactivity Monitoring Program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an UNRESTRICTED AREA, in the event of an uncontrolled release of the tanks contents. 01-03-LG

4.11.3 Deleted

4.11.4 Deleted

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Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. **Deletions** - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. **Additions** - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. **Modifications** - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. **Administrative** - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions -** The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions -** The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications -** The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative -** The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

Methodology For Mark-Up of Current TS
(Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers (1 Page)
Description of Changes (1 Page)

TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS

SECTION 3/4.11

Technical Specification Title	CHG. NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Liquid Holdup Tank	1	None	None	3.11.1	3.11.1.4
Explosive Gas Mixture	1	None	None	3.11.2	3.11.2.5
Gas Storage Tanks	1	None	None	3.11.3	3.11.2.6

DESCRIPTION OF CHANGES TO TS SECTION 3/4.11

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
1-01	LG	This TS will be moved into the improved TS, Administrative Controls, Section 5.5.12, Explosive Gas and Storage Tank Radioactivity Monitoring Program, consistent with NUREG-1431. The LCO, ACTION and SURVEILLANCE details of the TS will be incorporated into the program. The details are replaced with the requirement to limit liquid radwaste quantities per the appropriate version of 10CFR20 [].
1-02	LG	This TS will be moved into the improved TS, Administrative Controls, Section 5.5.12, Explosive Gas and Storage Tank Radioactivity Monitoring Program, consistent with NUREG-1431. The LCO, ACTION and SURVEILLANCE details of the TS will be incorporated into the program. The details are replaced with the requirement to comply with the system's design criteria.
1-03	LG	This TS will be moved into the improved TS, Administrative Controls, Section 5.5.12, Explosive Gas and Storage Tank Radioactivity Monitoring Program, consistent with NUREG-1431. The LCO, ACTION and SURVEILLANCE details of the TS will be incorporated into the program. The details are replaced with the requirement to comply with [Regulatory Guide 1.24 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressured Water Reactor Radioactive Gas Storage Tank Failure"].

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(1 page)

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.11

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
1-01 LG	The underlying requirements of this TS are now in ITS 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," which includes the requirement to limit liquid radwaste quantities per the appropriate version of 10CFR20 []. The details of the LCO, ACTION and SURVEILLANCE will be incorporated into the Explosive Gas and Storage Tank Radioactivity Monitoring Program.	Yes, moved to Explosive Gas Storage and Storage Tank Radioactivity Monitoring Program and ECG (See Attachment 21, page 32).	Yes, moved to Explosive Gas Storage and Storage Tank Radioactivity Monitoring Program.	No, not in CTS	No, not in CTS
1-02 LG	The underlying requirements of this TS are now in ITS 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," which includes the requirement to comply with the system's design criteria. The details of the LCO, ACTION and SURVEILLANCE will be incorporated into the Explosive Gas and Storage Tank Radioactivity Monitoring Program.	Yes, moved to Explosive Gas Storage and Storage Tank Radioactivity Monitoring Program and ECG (See Attachment 21, page 34).	Yes, moved to Explosive Gas Storage and Storage Tank Radioactivity Monitoring Program.	No, not in CTS	No, not in CTS
1-03 LG	The underlying requirements of this TS are now in ITS 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," which includes the requirement to comply with [RG 1.24]. The details of the LCO, ACTION and SURVEILLANCE will be incorporated into the Explosive Gas and Storage Tank Radioactivity Monitoring Program.	Yes, moved to Explosive Gas Storage and Storage Tank Radioactivity Monitoring Program and ECG (See Attachment 21, page 36).	Yes, moved to Explosive Gas Storage and Storage Tank Radioactivity Monitoring Program.	No, not in CTS	No, not in CTS

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

PAGE

I.	Organization	2
II.	Description of NSHC Evaluations.	3
III.	Generic NSHCs	
	"A" - Administrative Changes	5
	"R" - Relocated Technical Specifications	7
	"LG" - Less Restrictive (moving information out of the TS)	10
	"M" - More Restrictive	12
IV.	Specific NSHCs - "LS"	None

I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3...." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

Not Applicable

ENCLOSURE 5B

MARK-UP OF NUREG-1431 BASES

Not Applicable

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

None

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

None

JLS Conversion to Improved Technical Specifications

Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706270042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 5.0 - Design Features

ITS 4.0 - Design Features



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 5.0

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- ENCLOSURE 1 - CROSS-REFERENCE TABLES
- ENCLOSURE 2 - MARK-UP OF CURRENT TS
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- ENCLOSURE 4 - NO SIGNIFICANT HAZARDS CONSIDERATIONS
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- ENCLOSURE 5B - MARK-UP OF NUREG-1431 BASES
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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(1 Page)
METHODOLOGY	(3 Pages)

CROSS-REFERENCE TABLE FOR TS 5.0
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
5.1				4.1			
5.1.1			01-01-LG				
5.1.2			01-01-LG				
5.1.3			01-02-LG				
Fig 5.1-1			01-01-LG				
Fig 5.1-2			01-01-LG				
Fig 5.1-3			01-02-LG				
5.2			02-01-LG				
5.2.1			02-01-LG				
5.2.2			02-01-LG				
5.3				4.2			
5.3.1			03-01-A	4.2.1			
5.3.2			03-02-A	4.2.2			
5.4			04-01-LG				
5.4.1			04-01-LG				
5.4.2			04-01-LG				
5.5			05-01-LG				
5.5.1			05-01-LG				
5.6				4.3			
5.6.1			06-01-A	4.3.1			
5.6.1			06-01-A	4.3.1.1			
			06-02-M	4.3.1.2			
5.6.2				4.3.2			
5.6.3				4.3.3			
5.7			07-01-LG				
5.7.1			07-01-LG				
Tbl 5.7-1			07-01-LG				

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

Methodology for Cross-Reference Tables
(Continued)

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	-	The LCO operability requirement
APP	-	The APPLICABILITY requirement
CONDITION / ACTION	-	The ACTION requirements
SR	-	The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).

Note: The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is be used to cover all sub-paragraphs (e.g., "A.1" is be used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
5.1	5-1
5.2	5-1
5.3	5-5
5.4	5-5
5.5	5-6
5.6	5-6
5.7	5-6
Methodology	(2 Pages)

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

01-01-LG

~~5.1.1 The Exclusion Area shall be as shown in Figure 5.1 1.~~

LOW POPULATION ZONE

~~5.1.2 The Low Population Zone shall be as shown in Figure 5.1 2.~~

~~The DCCP site consists of approximately 750 acres which are adjacent to the Pacific Ocean in San Luis Obispo County, California, and is approximately twelve (12) miles west-southwest of the city of San Luis Obispo.~~

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

01-02-LG

~~5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figure 5.1 3.~~

~~The definition of UNRESTRICTED AREA used in implementing these Technical Specifications has been expanded over that in 10CFR20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10CFR100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10CFR50.36a.~~

5.2 CONTAINMENT

02-01-LG

CONFIGURATION

~~5.2.1 The containment is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:-~~

- ~~a. Nominal inside diameter = 140 feet.~~
- ~~b. Nominal inside height = 212 feet.~~
- ~~c. Minimum thickness of concrete walls = 3.6 feet.~~
- ~~d. Minimum thickness of concrete roof = 2.5 feet.~~
- ~~e. Minimum thickness of concrete floor pad = 14.5 feet.~~
- ~~f. Nominal thickness of steel liner, wall and dome = 3/8 inch.~~
- ~~g. Nominal thickness of steel liner, base = 1/4 inch.~~
- ~~h. Net free volume = 2.55×10^6 cubic feet.~~

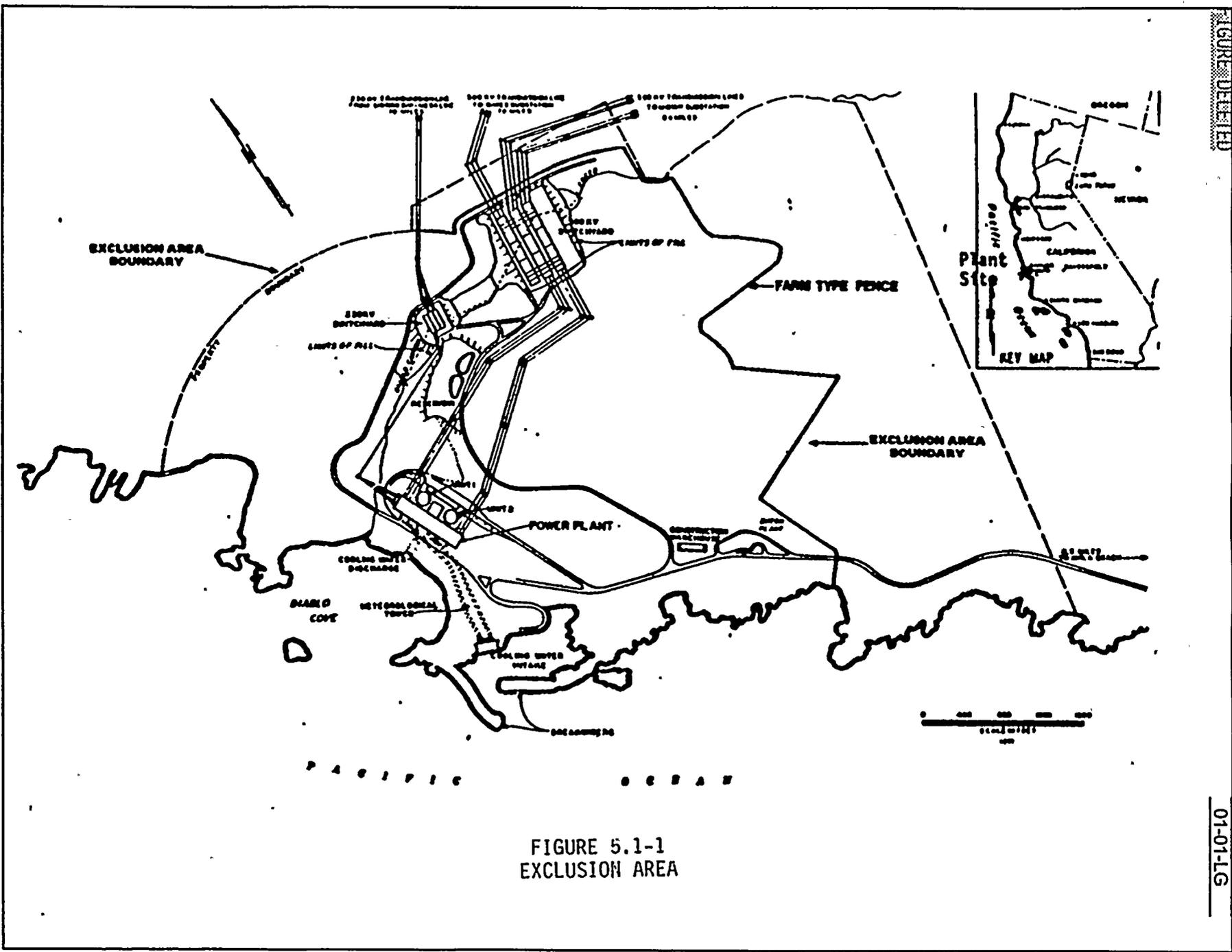


FIGURE 5.1-1
EXCLUSION AREA

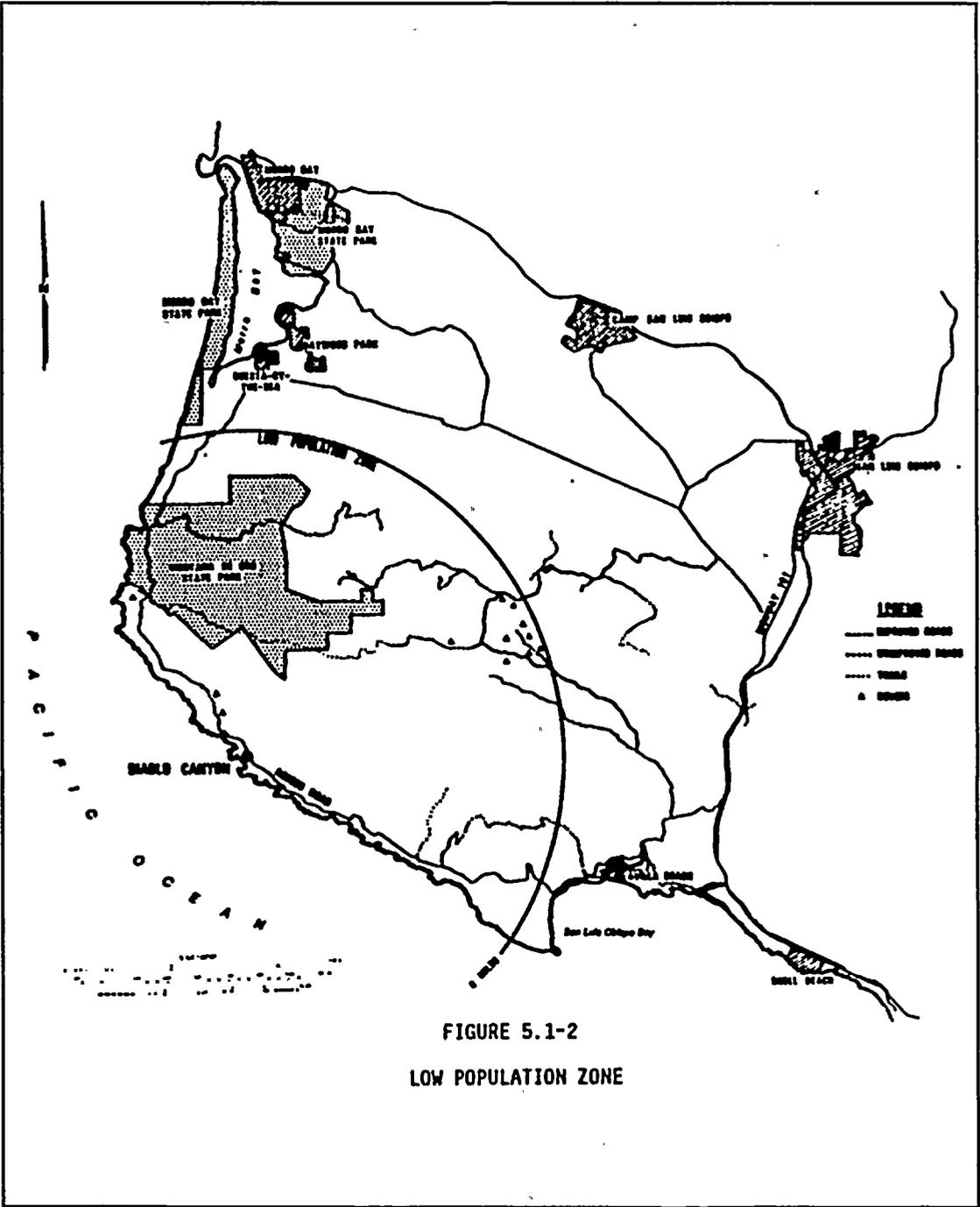


FIGURE 5.1-2
LOW POPULATION ZONE

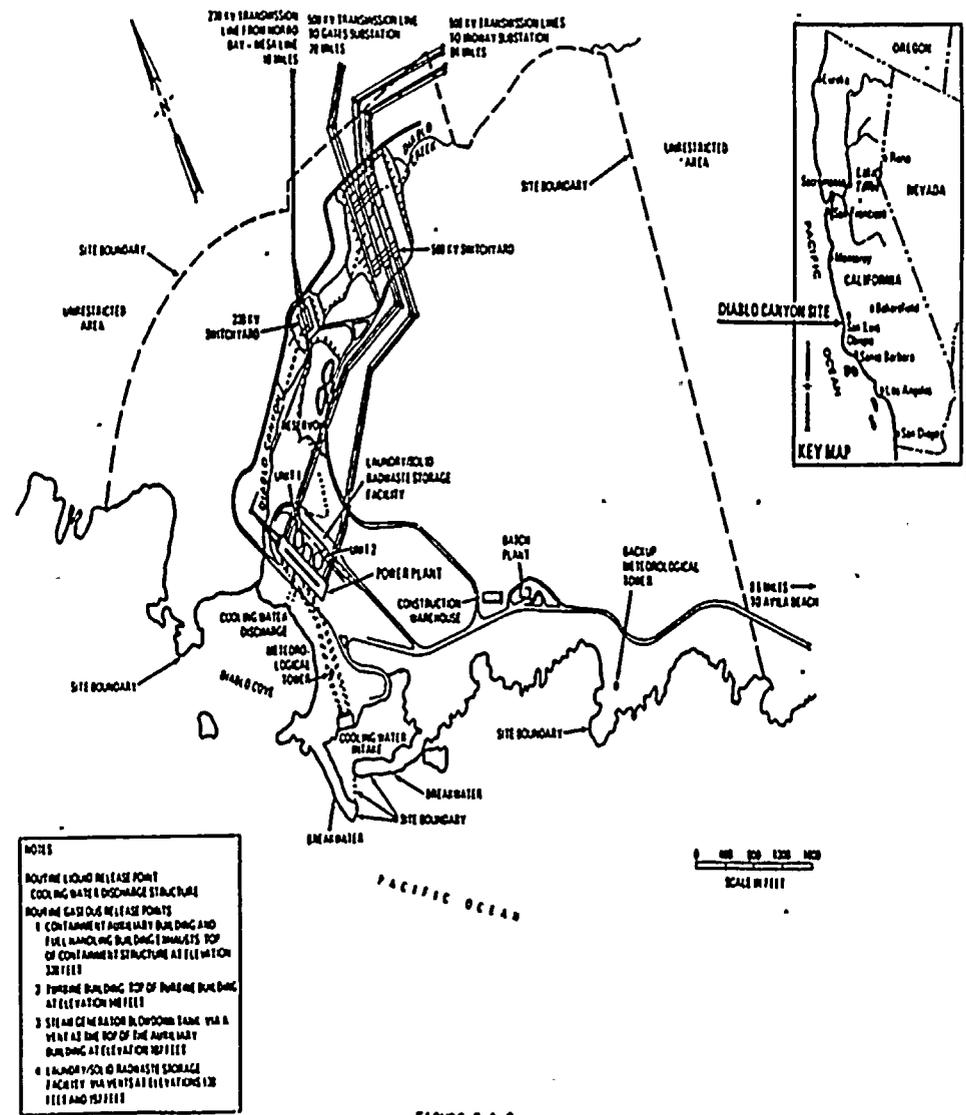


FIGURE 5.1-3
MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY
FOR RADIOACTIVE GASES AND LIQUID EFFLUENTS

DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE

02-01-LG

~~5.2.2 Containment is designed and shall be maintained for a maximum internal pressure of 47 psig and a temperature of 271°F, coincident with a Double Design Earthquake.~~

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy-4 or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO₂) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analysis to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core locations.

03-01-A

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% and cadmium as approved by the NRC. All control rods shall be clad with stainless steel tubing.

03-02-A

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

04-01-LG

- a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,
- b. For a pressure of 2485 psig, and
- c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,811 ± 100 cubic feet at a nominal T_{avg} of 576°F for Unit 1 and 12,903 ± 100 cubic feet at a nominal T_{avg} of 577°F for Unit 2.

04-01-LG

DESIGN FEATURES

5.5 METEOROLOGICAL TOWER LOCATION

05-01-LG

~~5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.~~

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

- a. ~~b.~~ A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties as described in Section 9.1 of the FSAR. 06-01-A
- b. ~~c.~~ A nominal 10.93 inch center-to-center distance between fuel assemblies placed in the storage racks.
- e. ~~a.~~ Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent.

~~New. The new fuel storage racks are designed and shall be maintained with:~~

06-02-M

- a. ~~Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent.~~
- b. ~~$k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1.1 of the FSAR.~~
- c. ~~$k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the FSAR, and~~
- d. ~~A nominal 22 inch center-to-center distance between fuel assemblies placed in the storage racks.~~

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 133.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

07-01-LG

~~5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.~~

TABLE 5.7.1
COMPONENT CYCLIC OR TRANSIENT LIMITS

07-01-LG

<u>COMPONENT</u>	<u>CYCLIC OR TRANSIENT LIMIT</u>	<u>DESIGN CYCLE OR TRANSIENT</u>
Reactor Coolant System	250 Heatup and Cooldown Cycles	200°F to 550°F to 200°F except for pressurizer, which is 200°F to 650°F to 200°F
	100 Loss of load cycles, with turbine trip and without immediate reactor trip	Above 15% RTP to 0% RTP
	50 cycles of loss of all offsite electrical power	Loss of all offsite power with turbine trip and initial power level at 100% RTP
	100 cycles of loss of flow in one reactor coolant loop	Loss of one reactor coolant pump with initial power level at 100% RTP
	500 reactor trip cycles	100% to 0% RTP
	12 inadvertent auxiliary spray actuation cycles	Spray water temperature differential > 320°F and < 560°F
	60 leak tests with no pressurization of the secondary side of steam generators	Pressurized to ≥ 2500 psig coincident
Secondary Coolant System	10 hydrostatic pressure tests	Pressurized to ≥ 3107 psig
	10 hydrostatic pressure tests each steam generator	Pressurized to ≥ 1356 psig coincident with the primary side at 0 psig
	10 Turbine roll tests rate > 100°F/hr	Turbine roll on RCP heat resulting in plant cooldown

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. Deletions - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. Additions - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. Modifications - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. Administrative - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

Methodology For Mark-Up of Current TS
(Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Description of Changes

(2 Pages)

DESCRIPTION OF CHANGES TO TS SECTION 5.0

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	LG	Figures representing site location/exclusion area boundary and low population zone replaced by text description to be consistent with NUREG-1431.
01-02	LG	Map for radioactive gaseous and liquid effluents moved to a licensee controlled document consistent with NUREG-1431.
02-01	LG	Containment design description moved to licensee controlled document consistent with NUREG-1431.
03-01	A	The description of the fuel assemblies was reworded to be consistent with NUREG-1431. Proposed rewording does not involve any technical changes.
03-02	A	Detailed information regarding control rod construction moved to a licensee controlled document. A reworded general description of the control rods is provided consistent with NUREG-1431.
03-03	A	Not Applicable to DCP. See Conversion Comparison Table. (Enclosure 3B)
04-01	LG	The description of the reactor coolant system volume and pressure and temperature limits is removed from the Technical Specifications (TS). This information is consistent with information already contained in the licensee controlled documents. The change is consistent with NUREG-1431.
05-01	LG	The meteorological tower location is removed from the TSs. This information is consistent with information already contained in licensee controlled documents. The change is consistent with NUREG-1431.
06-01	A	The fuel storage - criticality section is reformatted consistent with NUREG-1431. The proposed reformatting does not involve any technical changes.
06-02	M	The new fuel storage section is revised consistent with NUREG-1431. Proposed revisions provide details regarding analysis assumptions/limitations for the storage of new fuel (equivalent to that provided for spent fuel storage). The additional details are also consistent with licensee controlled documents.
06-03	A	Not Applicable to DCP. See Conversion Comparison Table. (Enclosure 3B)

DESCRIPTION OF CHANGES TO TS SECTION 5.0
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
07-01	LG	Component cyclic or transient limits table are removed from the TSs. A new component cyclic or transient limit program is added to section 5.5.5 of the improved TS, consistent with NUREG-1431, to assure that the requirements of the program continue to be controlled by the TS.

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(2 pages)

CONVERSION COMPARISON TABLE - CURRENT TS 5.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 LG	Figures representing the site location/exclusion area boundary and low population zone are moved to the FSAR and replaced by a text description.	Yes; already contained in FSAR Section 2.1.1, 2.1.2, and 2.1.3.3.	Yes; moved to FSAR	Yes; moved to USAR	Yes; moved to FSAR
01-02 LG	The map for radioactive gaseous and liquid effluents is moved to a licensee controlled document .	Yes; already contained in FSAR.	Yes; moved to FSAR	Yes; moved to USAR	Yes; moved to FSAR
02-01 LG	The containment design description is moved to a licensee controlled document.	Yes; already contained in FSAR section 3.8.1.1.	Yes; moved to FSAR	Yes; moved to USAR	Yes; moved to FSAR
03-01 A	The description of the fuel assemblies was reworded. Proposed rewording does not involve any technical changes.	Yes	No; see 03-03-A	No; see 03-03-A	No; see 03-03-A
03-02 LG	Detailed information regarding control rod construction moved to a licensee controlled document. A reworded general description of the control rods is provided.	Yes; moved to FSAR	Yes; moved to FSAR	Yes; moved to USAR	Yes; moved to FSAR
03-03 A	The description of the core is generalized and reference to the initial core is deleted. A requirement is added which requires that fuel assembly designs be analyzed using NRC approved methodologies.	No; see 03-01-A	Yes	Yes	Yes
04-01 LG	The description of the reactor coolant system volume and pressure and temperature limits is moved from the Technical Specifications (TS) to a licensee controlled document.	Yes; moved to FSAR	Yes; moved to FSAR	Yes; moved to USAR	Yes; moved to FSAR
5-01 LG	The meteorological tower location is moved from the TS to a licensee controlled document.	Yes; moved to FSAR	Yes; moved to FSAR	Yes; moved to USAR	Yes; moved to FSAR
6-01 A	The fuel storage - criticality section is reformatted. The proposed reformatting does not involve any technical changes.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 5.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
6-02 M	The new fuel storage section is revised to provide details regarding analysis assumptions/limitations for the storage of new fuel (equivalent to that provided for spent fuel storage). The additional details are also consistent with licensee controlled documents.	Yes	No; part of CTS	Yes	Yes
6-03 A	The referenced section of the Wolf Creek Updated Safety Analysis Report has been revised to reference the appropriate section.	No	No	Yes	No
7-01 LG	Component cyclic or transient limits table are moved from the TS to a licensee controlled document. A new component cyclic or transient limit program is added to section 5.5.5 of the improved TS.	Yes; moved to FSAR	Yes; moved to FSAR	Yes; moved to USAR	Yes; moved to FSAR

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

PAGE

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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3..." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
10CFR50.92 EVALUATION
FOR
ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

- 1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
- 2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
- 3. Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION BASES,
FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
4.1	4.0-1
4.2	4.0-1
4.3	4.0-1

Methodology

(2 Pages)

4.0 DESIGN FEATURES

4.1 Site Location ~~[Text description of site location.]~~

B-PS

The DCPP site consists of approximately 750 acres which are adjacent to the Pacific Ocean in San Luis Obispo County, California, and is approximately twelve (12) miles west-southwest of the city of San Luis Obispo.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain ~~[157]~~ 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO₂) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions ~~locations~~.

B-PS

ED

4.2.2 ~~[Control Rod]~~ Assemblies

The reactor core shall contain ~~[48]~~ 53 control rod assemblies. The control rod material shall be ~~[silver, indium, and cadmium, boron carbide, or hafnium metal]~~ as approved by the NRC.

B-PS

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 5.0 weight percent;
- b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 ~~[2.3]~~ of the FSAR;

B-PS

B-PS

4.3 Fuel Storage (continued)

- c. A nominal ~~[9.15]~~ 10 inch center to center distance between fuel assemblies placed in ~~the high density fuel storage racks~~ B-PS
B-PS
- ~~[d. A nominal [10.95] inch center to center distance between fuel assemblies placed in [low density fuel storage racks].]~~
- ~~[e. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure [3.7.17-1] may be allowed unrestricted storage in [either] fuel storage rack(s); and]~~ B-PS
- ~~[f. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure [3.7.17-1] will be stored in compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]~~

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of ~~[4.5]~~ 5.0 weight percent; B-PS
- b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in ~~[Section 9.1.1]~~ of the FSAR; B-PS
- c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in ~~[Section 9.1.1]~~ of the FSAR; and B-PS
- d. A nominal ~~[10.95]~~ 22 inch center to center distance between fuel assemblies placed in the storage racks. B-PS

4.3.2 Drainage

The spent fuel storage pool ~~is~~ are designed and shall be maintained to prevent inadvertent draining of the pool below elevation ~~[23]~~ 133 ft. B-PS
B-PS

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~[1737]~~ 1324 fuel assemblies. B-PS

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes- Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.
 2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.

**Methodology For Mark-up of NUREG-1431 Specifications
(Continued)**

3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

Not Applicable

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(1 Page)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 4.0

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

CHANGE
NUMBER

JUSTIFICATION

4.0-1

Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(1 Page)

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 4.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
4.0-1	Revise Section 4.2.1 to maintain the option for the limited use of vacancies for fuel rod substitutions as reflected in the current licensing basis.	No; Diablo Canyon does not use vacancies.	No	No; Wolf Creek does not use vacancies.	Yes

JLS Conversion to Improved Technical Specifications Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

CTS 6.0 - Administrative Controls

ITS 5.0 - Administrative Controls



IMPROVED TECHNICAL SPECIFICATIONS CONVERSION

CURRENT TS SECTION 6.0

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ENCLOSURE 1

CROSS-REFERENCE TABLES

CONVERSION CROSS-REFERENCE CONTENTS

CONVERSION TABLE SORTED BY CURRENT TS	(4 Pages)
CONVERSION TABLE SORTED BY IMPROVED TS	(4 Pages)
METHODOLOGY	(3 Pages)

CROSS-REFERENCE TABLE FOR 6.0
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
6.1				5.1			
6.1.1				5.1.1			
6.1.2			01-01-A	5.1.2			
6.2				5.2			
6.2.1				5.2.1			
6.2.1		a		5.2.1		a	
6.2.1		b		5.2.1		b	
6.2.1		c		5.2.1		c	
6.2.1		d		5.2.1		d	
6.2.2			01-02-A	5.2.2			
6.2.2		a	01-02-A	5.2.2		a	5.2-1
6.2.2		b	01-05-A	5.2.2			5.2-2
Table 6.2-1			01-06-LG	5.2.2		b	
Table 6.2-1		note	01-03-A			not used	
6.2.2		c		5.2.2		c	
6.2.2		d	01-03-A			not used	
6.2.2		f		5.2.2		d	
6.2.2		g		5.2.2		e	
6.2.4				5.2.2		f	5.2-6
6.3				5.3			
6.3				5.3.1			
6.4				5.3.1			5.3-1
6.5							
6.6		a	02-01-A			not used	
6.6		b	02-15-LG				
6.7			02-02-LS4			not used	
6.8				5.4			
6.8.1				5.4.1			
6.8.1		a		5.4.1		a	

CROSS-REFERENCE TABLE FOR 6.0
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
6.8.1		b		5.4.1		b	
6.8.1		c					
6.8.1		d					
6.8.1		e	02-03-A			not used	
6.8.1		f	02-03-A			not used	
6.8.1		g		5.4.1		c	
6.8.1		h		5.4.1		d	
6.8.1		(new)	02-08-M	5.4.1		e	
6.8.2							
6.8.3							
6.8.4				5.5			
6.8.4		a		5.5.2			
6.8.4		b	02-04-LG			relocated	
6.8.4		c		5.5.10			
6.8.4		d		5.5.17		new	5.5-6
6.8.4		e		5.5.3			5.5-2
6.8.4		f		5.5.18		new	5.5-6
6.8.4		g	02-09-A 02-05-A 02-14-M 02-07-A	5.5.4			5.5-1 5.5-13
6.8.4		h	02-06-A			not used	
6.8.4		i	02-17-LS1	5.5.7			5.5-14
				5.5.6		NA	
				5.5.5			
				5.5.8			5.5-3
				5.5.9			5.5-8
				5.5.11			
				5.5.12			5.5-9
				5.5.13			5.5-5 5.5-8
6.8.4		(new)		5.5.14			

CROSS-REFERENCE TABLE FOR 6.0
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
6.8.4		(new)		5.5.15			
6.8.4		j		5.5.16			5.5-4
6.9			03-01-A	5.6			
6.9.1			03-01-A	5.6			
6.9.1.1			03-02-A			not used	
6.9.1.2			03-02-A			not used	
6.9.1.3			03-02-A			not used	
6.9.1.4			03-03-A 03-04-A	5.6.1			5.6-4
6.9.1.5			03-03-A 03-07-A 02-09-A	5.6.2			5.6-3
6.9.1.6			03-03-A 03-06-A 02-09-A	5.6.3			5.6-4
6.9.1.7			03-08-A	5.6.4			
6.9.1.8		a	03-14-M 03-15-M	5.6.5		a	
6.9.1.8		b		5.6.5		b	
6.9.1.8		c		5.6.5		c	
6.9.1.8		d	03-08-A	5.6.5		d	
		new	03-13-M	5.6.6			5.6-5
				5.6.7		NA	5.6-2
		new	03-13-M	5.6.8			
				5.6.9		NA	
				5.6.10			
6.9.2			03-08-A			deleted	
6.10			03-09-LG			relocated	
6.11			03-10-LG			relocated	
6.12				5.7			
6.12.1			03-11-A	5.7.1			5.7-1

CROSS-REFERENCE TABLE FOR 6.0
Sorted by Current TS

Current TS				Improved TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
6.12.2			03-11-A 03-19-A 03-20-LS3	5.7.2			5.7-1 5.7-2
6.12.2			03-11-A 03-20-LS3	5.7.3			5.7-1
6.13			03-12-LG			relocated	
6.14				5.5.1			
6.14.1			02-09-A	5.5.1		a	
6.14.1			02-09-A	5.5.1		b	
6.14.2		a	03-09-LG 02-09-A	5.5.1		a	
6.14.2		b	02-13-LG	5.5.1		b	
6.14.2		c	02-09-A 03-06-A	5.5.1		c	

CROSS-REFERENCE TABLE FOR 6.0
Sorted by Improved TS

Improved TS				Current TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
5.1				6.1			
5.1.1				6.1.1			
5.1.2				6.1.2			01-01-A
5.2				6.2			
5.2.1				6.2.1			
5.2.1		a		6.2.1		a	
5.2.1		b		6.2.1		b	
5.2.1		c		6.2.1		c	
5.2.1		d		6.2.1		d	
5.2.2				6.2.2			01-02-A
5.2.2		a	5.2-1	6.2.2		a	01-02-A
5.2.2			5.2-2	6.2.2		b	01-05-A
5.2.2		b		Table 6.2-1			01-06-LG
		not used		Table 6.2-1		note	01-03-A
5.2.2		c		6.2.2		c	
		not used		6.2.2		d	01-03-A
5.2.2		d		6.2.2		f	
5.2.2		e		6.2.2		g	
5.2.2		f	5.2-6	6.2.4			
5.3				6.3			
5.3.1				6.3			
5.3.1			5.3-1	6.4			
		not used		6.5			
		not used		6.6		a	02-01-A
				6.6		b	02-15-LG
		not used		6.7			02-02-LS4
5.4				6.8			
5.4.1				6.8.1			
5.4.1		a		6.8.1		a	

CROSS-REFERENCE TABLE FOR 6.0
Sorted by Improved TS

Improved TS				Current TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
5.4.1		b		6.8.1		b	
				6.8.1		c	
				6.8.1		d	
		not used		6.8.1		e	02-03-A
		not used		6.8.1		f	02-03-A
5.4.1		c		6.8.1		g	
5.4.1		d		6.8.1		h	
5.4.1		e		6.8.1		(new)	02-08-M
				6.8.2			
				6.8.3			
5.5				6.8.4			
5.5.1				6.14			
5.5.1		a		6.14.1			02-09-A
5.5.1		b		6.14.1			02-09-A
5.5.1		a		6.14.2		a	03-09-LG 02-09-A
5.5.1		b		6.14.2		b	02-13-LG
5.5.1		c		6.14.2		c	02-09-A 03-06-A
5.5.2				6.8.4		a	
5.5.3			5.5-2	6.8.4		e	
		relocated		6.8.4		b	02-04-LG
5.5.4			5.5-1 5.5-13	6.8.4		g	02-09-A 02-05-A 02-14-M 02-07-A
5.5.5							
5.5.6		NA					
5.5.7			5.5-14	6.8.4		l	02-17-LS1
5.5.8			5.5-3				
5.5.9			5.5-8				
5.5.10				6.8.4		c	

CROSS-REFERENCE TABLE FOR 6.0
Sorted by Improved TS

Improved TS				Current TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
5.5.11							
5.5.12			5.5-9				
5.5.13			5.5-5 5.5-8				
5.5.14				6.8.4		(new)	
5.5.15				6.8.4		(new)	
5.5.16			5.5-4	6.8.4		j	
5.5.17		new	5.5-6	6.8.4		d	
5.5.18		new	5.5-6	6.8.4		f	
		not used		6.8.4		h	02-06-A
5.6				6.9			03-01-A
5.6				6.9.1			03-01-A
		not used		6.9.1.1			03-02-A
		not used		6.9.1.2			03-02-A
		not used		6.9.1.3			03-02-A
5.6.1			5.6-4	6.9.1.4			03-03-A 03-04-A
5.6.2			5.6-3	6.9.1.5			03-03-A 03-07-A 02-09-A
5.6.3			5.6-4	6.9.1.6			03-03-A 03-06-A 02-09-A
5.6.4				6.9.1.7			03-08-A
5.6.5		a		6.9.1.8		a	03-14-M 03-15-M
5.6.5		b		6.9.1.8		b	
5.6.5		c		6.9.1.8		c	
5.6.5		d		6.9.1.8		d	03-08-A
5.6.6			5.6-5			new	03-13-M
5.6.7		NA	5.6-2				
5.6.8						new	03-13-M
5.6.9		NA					

CROSS-REFERENCE TABLE FOR 6.0
Sorted by Improved TS

Improved TS				Current TS			
Item	Code	Para.	Change	Item	Code	Para.	Change
5.6.10							
		deleted		6.9.2			03-08-A
		relocated		6.10			03-09-LG
		relocated		6.11			03-10-LG
5.7				6.12			
5.7.1			5.7-1	6.12.1			03-11-A
5.7.2			5.7-1 5.7-2	6.12.2			03-11-A 03-19-A 03-20-LS3
5.7.3			5.7-1	6.12.2			03-11-A 03-20-LS3
		relocated		6.13			03-12-LG

Methodology for Cross-Reference Tables

The cross-reference tables provide a guide to location of all current TS LCOs, ACTIONS, Surveillances, Tables, and Figures in the improved TS. It also includes the location of items that have been located out of the improved TS.

The cross-reference table contains the following columns:

Current TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated technical specification.

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general, the numbering and lettering used in the current TS will be provided but in some cases it may be appropriate to provide a description. For example in specification 3/4.7.7.1, the actions are arranged by those that apply in MODES 1, 2, 3, & 4 and those that apply in MODES 5, 6 and during movement of irradiated fuel assemblies. Appropriate entries in this column for these respective actions might be "MODES 1-4" and "MODES 5, 6, etc." Multiple paragraphs are not listed in the same row (e.g., "a and b").

New This item has been added to reflect a requirement in NUREG-1431 that is not addressed in the current TS.

NA This item is not in the current TS because it does not apply.

Note: When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry is made for each cross-reference. A single entry is not used to identify the multiple paragraphs in the improved TS. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, are made for each such paragraph in the current TS.

**Methodology for Cross-Reference Tables
(Continued)**

Improved TS:

LCO/SR number (Item) -

This column lists the LCO or SR number which applies as listed in the associated specification or uses the following code:

Relocated	This item is relocated to another licensee control document outside the TS (see Code for specific reference location).
-----------	--

Requirement code (Code) -

This column identifies the portion of the specification affected using the following code:

LCO	- The LCO operability requirement
APP	- The APPLICABILITY requirement
CONDITION / ACTION	- The ACTION requirements
SR	- The SURVEILLANCE REQUIREMENTS

In addition, specific plant document acronyms are used to list the licensee controlled documents where the item will be relocated to (e.g., FSAR, TRM, etc.).

Note: The applicability of a current specification is assumed to transfer to the same improved specification as the LCO. The cross-reference for the applicability for the specification is only identified in the table by a separate entry if the cross-reference is not clear (e.g., several current specifications with different applicability are moved into the same specification in the improved TS, or a footnote in the applicability of the current TS is moved to a different portion of the specification in the improved TS).

Paragraph (Para) -

This column identifies the affected paragraph. In general the numbering and lettering used in the improved TS is provided but in some cases it may be appropriate to provide a description.

New	This item has been added to the improved TS and was not addressed in the NUREG-1431.
Not Used	This item will not be used in the improved TS, nor relocated to another document (e.g., requirements already adequately addressed by regulations).
NA	This item from NUREG-1431 is not included in the improved TS because it does not apply (e.g., specification unique to Ice Condenser Containments).
Note:	The paragraph is only identified to the extent necessary to adequately describe the cross-reference. For example, if the cross-reference applies to the entire condition, it is appropriate to list the "Requirement Code" as "CONDITION" and the "Paragraph" as "A." If the correct cross-reference is only to the required action, an appropriate cross-reference would be to "Requirement Code" as "ACT" and "Paragraph" as "A.1."

Methodology for Cross-Reference Tables
(Continued)

Note:

When a single paragraph in the current TS crosses to multiple locations in the improved TS, a new entry for each cross-reference is made. Since multiple paragraphs in the current TS may cross-reference to the same paragraph in the improved TS, separate entries, each referencing the same location in the improved TS, is made for each such paragraph in the current TS. Multiple paragraphs are not listed (e.g. "A.1.1 and A.1.2") although a "higher tier" number is used to cover all sub-paragraphs (e.g., "A.1" is used to identify all subparagraphs such as A.1.1, A.1.2, etc.).

ENCLOSURE 2

MARK-UP OF CURRENT TS

Mark-up

<u>SPECIFICATION</u>	<u>PAGE</u>
6.1	6-1
6.2	6-1
6.3	6-6
6.4	6-7
6.5	N/A
6.6	6-7
6.7	6-7
6.8	6-7
6.9	6-16
6.10	6-19
6.11	6-21
6.12	6-21
6.13	6-22
6.14	6-23
Methodology	(2 Pages)

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Vice President, Diablo Canyon Operations and Plant Manager, hereinafter called Plant Manager, shall be responsible for overall plant operation and shall delegate in writing the succession to this responsibility during his absence. The Plant Manager or his designee shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affects nuclear safety.

6.1.2 The Shift Foreman (or during his absence from the Control Room, a designated individual, other than the Shift Technical Advisor with an active SRO license) shall be responsible for the Control Room Command function while either unit is in MODES 1, 2, 3, or 4. While both units are in MODE 5 or 6, an individual with an active Senior Reactor Operator (SRO) or Reactor Operator (RO) license shall be designated to assume the control room command function. ~~A management directive to this effect signed by the Senior Vice President and General Manager, Nuclear Power Generation shall be reissued to all plant personnel on an annual basis.~~

01-01-A

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for plant operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

- a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR Update.
- b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.
- c. The Senior Vice President and General Manager - Nuclear Power Generation shall have corporate responsibilities for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.
- d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.0 ADMINISTRATIVE CONTROLS

6.2.2 PLANT UNIT STAFF

~~The unit staff organization shall include the following:~~

01-02-A

- a. ~~Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6-2-1. A non-licensed operator shall be assigned to each reactor containing fuel with a total of three non-licensed operators required for both units.~~
- b. ~~At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Operator shall be in the Control Room.~~
- c. A Health Physics Technician* shall be on site when fuel is in the reactor;

01-05-A

*The Health Physics Technician position may be unfilled for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the position.

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TABLE 6.2-1

01-06-LG

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, OR 4	BOTH UNITS IN MODE 5 OR 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3 OR 4 AND ONE UNIT IN MODE 5 OR 6 OR DEFUELED
	SFM	1	1
SOL	1	none ^b	1
OL	3a	2a	3a
AO	3a	3a	3a
STA	1*	none	1*

- ~~_____~~ SFM - Shift Foreman with a Senior Operator License
- ~~_____~~ SOL - Individual with a Senior Operator License
- ~~_____~~ OL - Individual with an Operator License
- ~~_____~~ AO - Auxiliary Operator
- ~~_____~~ STA - Shift Technical Advisor

The Shift Crew Composition may be one less than the minimum requirements of ~~Table 6.2-1~~ 10CFR 50.54(m)(2)(i) and 6.2.2 a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of ~~Table 6.2-1~~. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

01-06-LG

During any absence of the Shift Foreman from the control room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Foreman from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the control room command function.

~~a/At least one of the required individuals must be assigned to the designated position for each unit.~~

01-03-A

~~b/At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during Core Alterations on either unit, who has no other concurrent responsibilities.~~

01-06-LG

*The STA position shall be manned in MODES 1, 2, 3, and 4 unless an individual with a Senior Operator license meets the Commission Policy Statement on Engineering Expertise on Shift.

ADMINISTRATIVE CONTROLS

PLANT STAFF (Continued)

01-03-A

- d. ~~All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.~~
- e. Deleted.
- f. Administrative procedures shall be developed and implemented to limit the working hours of plant staff who perform safety-related functions; e.g., licensed Senior Operators, licensed Operators, Health Physicists, auxiliary operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

- 1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;
- 2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time;
- 3) A break of at least 8 hours should be allowed between work periods, including shift turnover time; and
- 4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager or his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized; and

- g. The Operations Manager shall hold a senior reactor operator license.

ADMINISTRATIVE CONTROLS

6.2.3 DELETED

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4 The Shift Technical Advisor, shall provide advisory technical support to the Shift Foreman in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the plant.

6.3 PLANT STAFF QUALIFICATIONS

6.3 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions, except for the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 2, April 1987 for Radiation Protection Manager. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of Part 55 and the supplemental requirements specified in Section A of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

ADMINISTRATIVE CONTROLS

6.4 TRAINING

6.4 A retraining and replacement training program for the plant staff shall be maintained under the direction of a designated member of the facility staff and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10CFR Part 55.

6.5 DELETED

6.6 REPORTABLE EVENT ACTION

~~6.6 The following actions shall be taken for REPORTABLE EVENTS:~~

02-01-A

~~a. The Commission shall be notified and a report submitted pursuant to the requirements of 10CFR 50.73; and~~

~~b. Each REPORTABLE EVENT shall be reviewed by the PSRC and the results of this review submitted to NSOC and the Senior Vice President and General Manager, Nuclear Power Generation.~~

02-15-LG

6.7 SAFETY LIMIT VIOLATION

~~6.7 The following actions shall be taken in the event a Safety Limit is violated:~~

02-02-LS4

~~a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President and General Manager, Nuclear Power Generation and NSOC shall be notified within 24 hours;~~

~~b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems or structures, and (3) corrective action taken to prevent recurrence;~~

~~c. The Safety Limit Violation Report shall be submitted to the Commission, NSOC and the Senior Vice President and General Manager, Nuclear Power Generation within 14 days of the violation; and~~

~~d. Critical operation of the unit shall not be resumed until authorized by the Commission.~~

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

- a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

ADMINISTRATIVE CONTROLS

6.8 PROCEDURES AND PROGRAMS (Continued)

- b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;
- c. Deleted;
- d. Deleted;
- e. ~~PROCESS CONTROL PROGRAM implementation;~~
- f. ~~ODCP and ERMP implementation; and~~
- g. Quality Assurance Program for effluent and environmental monitoring; and
- h. Fire Protection Program Implementation.

02-03-A

~~(new) All programs specified in Specification 6.8.4~~

02-08-M

6.8.2 Deleted.

6.8.3 Deleted.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Reactor Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include portions of the Recirculation Spray System, Safety Injection System, Chemical and Volume Control System, Residual Heat Removal System, RCS Sample System, and Liquid and Gaseous Radwaste Treatment Systems. The program shall include the following:

- 1) Preventive maintenance and periodic visual inspection requirements, and
- 2) Integrated leak test requirements for each system at refueling cycle intervals or less.

~~b. In-Plant Radiation Monitoring~~

02-04-LG

~~A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:~~

- ~~1) Training of personnel,~~
- ~~2) Procedures for monitoring, and~~
- ~~3) Provisions for maintenance of sampling and analysis equipment.~~

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

- 1) Identification of a sampling schedule for the critical variables and control points for these variables.
- 2) Identification of the procedures used to measure the values of the critical variables.
- 3) Identification of process sampling points, including monitoring the discharge of the condensate pumps for evidence of condenser in-leakage.
- 4) Procedures for the recording and management of data.
- 5) Procedures defining corrective actions for all off-control point chemistry conditions, and
- 6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

- 1) Training of personnel, and
- 2) Procedures for monitoring.

e. Postaccident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

- 1) Training of personnel.
- 2) Procedures for sampling and analysis, and
- 3) Provisions for maintenance of sampling and analysis equipment.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

f. Containment Polar and Turbine Building Cranes

A program which will ensure that: 1) the position of the containment polar cranes precludes jet impingement from a postulated pipe rupture; and 2) the operation of the turbine building cranes is consistent with the restrictions associated with the current Hosgri seismic analysis of the turbine building. This program shall include the following:

- 1) Training of personnel, and
- 2) Procedures for the containment polar and turbine building cranes operation.

g. Radioactive Effluent Controls Program

A program shall be provided conforming with 10CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the RMCP ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- | | |
|--|--|
| 1) Limitations on the operability functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance requirements and setpoint determination in accordance with the methodology in the ODCP ODCM. | <u>02-09-A</u> |
| 2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10CFR 20.1001-20.2402, 10CFR Part 20, Appendix B, Table II, Column 2. | <u>02-05-A</u>
<u>02-09-A</u>
<u>02-14-M</u> |
| 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10CFR 20.1302 and with the methodology and parameters in the ODCP ODCM. | <u>02-09-A</u> |
| 4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREA conforming to Appendix I to 10CFR Part 50. | <u>02-07-A</u> |
| 5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCP ODCM at least every 31 days
Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with methodology and parameters in the ODCM. Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least every 31 days. | <u>02-07-A</u> |

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

02-05-A

- 6) Limitations on the operability ~~functional capability~~ and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31 - day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50.
- 7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY shall be limited to the following:
 - (1) For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and
 - (2) For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ.
- 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10CFR part 50.
- 9) Limitations on the annual and quarterly doses to MEMBERS OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and
- 10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

h. ~~Radiological Environmental Monitoring Program~~

02-06-A

~~A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the RMCP, (2) conform to the guidance of Appendix I to 10CFR Part 50, and (3) include the following:~~

- ~~1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ERMP.~~

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

h. Radiological Environmental Monitoring Program

02-06-A

- ~~2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and~~
- ~~3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in the environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.~~

i. Reactor Coolant Pump Flywheel Inspection

Inspect each reactor coolant pump flywheel in accordance with the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

~~In lieu of Position C.4.b(1) and C.4.b(2), conduct a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or conduct a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels once every ten years coinciding with the Inservice Inspection schedule as required by ASME Section XI.~~

02-17-LS1

~~-----Reviewer's Note-----~~

- ~~1. Licensees shall confirm that the flywheels are made of SA 533 B material. Further, licensee having Group 15 flywheels (as determined in WCAP-14535 "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination") need to demonstrate that material properties of their A516 material is equivalent to SA 533 B material, and its reference temperature, RT_{10Y} , is less than 30°F.~~
- ~~2. For flywheels not made of SA 533 B or A516 material, licensees need to either demonstrate that the flywheel material properties are bounded by those of SA 533 B material, or provide the minimum specified ultimate tensile stress, the fracture toughness, and the reference temperature, RT_{10Y} , for that material. For the latter, the licensee should employ these material properties, and use the methodology in the topical report, as extended in the two responses to the staff's RAI, to provide an assessment to justify a change in inspection schedule for their plants.~~

j. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10CFR 50.54(o) and 10CFR 50.

Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995."

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 47 psig.

The maximum allowable containment leakage rate, L_a , at P_a , shall be 0.10 % of containment air weight per day.

Leakage rate acceptance criteria are:

- a. Containment overall leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60 L_a$ for the Type B and Type C tests and $\leq 0.75 L_a$ for Type A tests;
- b. Air lock testing acceptance criteria are:
 - 1) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$.
 - 2) For each door, leakage rate is $\leq 0.01 L_a$ when pressurized to ≥ 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

02-11-M

NEW PROGRAM INSERTS	
INSERT NUMBER	PROGRAM NAME
10	Technical Specification Bases Control Program
11	Safety Function Determination Program

Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications:

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 1. a change in the TS incorporated in the license, or
 2. a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- d. Proposed changes that meet the criteria of b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0-6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0-6. The SFDP shall contain the following:

- a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
- b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;
- c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
- d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

- a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
- b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

03-01-A

6.9 REPORTING REQUIREMENTS

~~The following reports shall be submitted in accordance with 10CFR 50.4~~
ROUTINE REPORTS

~~6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC in accordance with 10CFR 50.4.~~

STARTUP REPORTS

03-02-A

~~6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.~~

~~6.9.1.2 The initial Startup Report shall address each of the startup tests identified in Chapter 14 of the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate acceptability of the change and/or modification.~~

~~6.9.1.3 Startup Reports shall be submitted within: (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.~~

ANNUAL REPORTS*

03-03-A

~~6.9.1.4 Annual Reports covering the activities of the unit as described below during the previous calendar year shall be submitted prior to March 31 of each year. The initial report shall be submitted prior to March 31 of the year following initial criticality.~~

Occupational Radiation Exposure Report

~~Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man~~

*A single submittal may be made for a multiple-unit plant. The submittal should combine those sections that are common to all units at the plant.

ADMINISTRATIVE CONTROLS

ANNUAL REPORTS (Continued)

rem exposure for whom monitoring was performed, receiving an annual deep dose equivalent \geq 100 mrems and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket ionization chamber, thermoluminescence (TLD), electronic dosimeter or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body deep dose equivalent received from external sources shall be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

03-03-A

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8 will be included in the annual report. The following information shall be included: (1) reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) clean up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of specific activity above the steady state level; and (5) the time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

03-04-A

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT**

6.9.1.5 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the RMCP ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10CFR Part 50.

02-09-A

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

03-07-A

*This tabulation supplements the requirements of 10CFR 20.407 20.2206.
**A single submittal may be made for a multiple unit plant.

03-03-A

ADMINISTRATIVE CONTROLS

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.6 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted before prior to May 1 of each year in accordance with 10CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the RMCP ODCM and PCP, (2) in conformance with 10CFR 50.36a and Section IV.B.1 Appendix I to 10CFR Part 50.

03-06-A

02-09-A

MONTHLY OPERATING REPORT

6.9.1.7 Routine reports of operating statistics and shutdown experience, including documentation of all challenges and failures to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC in accordance with 10CFR 50.4, no later than the 15th of each month following the calendar month covered by the report.

03-08-A

CORE OPERATING LIMITS REPORT

6.9.1.8.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Shutdown Rod Insertion Limits for Specification 3/4.1.3.5.
2. Control Rod Insertion Limits for Specification 3/4.1.3.6.
3. Axial Flux Difference for Specification 3/4.2.1.
4. Heat Flux Hot Channel Factor, $K(Z)$ and $W(Z) - F_Q(z)$ (F_Q for Specification 3/4.2.2), and ^{RTP}
5. RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^N$ ($F_{\Delta H}$ and $PF_{\Delta H}$ for Specification 3/4.2.3)

6. SHUTDOWN MARGIN for Specifications 3/4.1.1.1, 3/4.1.1.2, and 3/4.10.1.

03-14-M

7. Moderator Temperature Coefficient Limits in Specification 3/4.1.1.3, and

8. Refueling Boron Concentration Limits in Specification 3/4.9.1

03-15-M

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, Revision 1A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, February 1994 (Westinghouse Proprietary).
2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (Westinghouse Proprietary).

*A single submittal may be made for a multiple unit plant. The submittal should shall combine those sections that are common to all units at the plant; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

03-03-A

ADMINISTRATIVE CONTROLS

- 3. WCAP-8385, Power Distribution Control and Load Following Procedures, September 1974 (Westinghouse Proprietary),
 - 4. WCAP-10054-P-A, Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code, August 1985. (Westinghouse Proprietary), and
 - 5. WCAP-10266-P-A, Revision 2 with Addenda, The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, December 14, 1987. (Westinghouse Proprietary).
- c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.
- d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided, upon issuance for each reload cycle, to the NRC. ~~Document Control Desk, with copies to the Regional Administrator and Resident Inspector.~~ 03-08-A

SPECIAL REPORTS

~~6.9.2 Special reports shall be submitted to the NRC in accordance with 10CFR 50.4 within the time period specified for each report.~~

03-08-A

6.10 RECORD RETENTION

03-09-LG

~~In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.~~

~~6.10.1 The following records shall be retained for at least 5 years:~~

- ~~a. Records and logs of unit operation covering time interval at each power level;~~
- ~~b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;~~
- ~~c. ALL REPORTABLE EVENTS;~~

New Report Inserts	
Insert Number	Report Name
21	RCS Pressure and Temperature Limits Report(PTLR)
22	PAM Report

Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3/4-4-9 (ITS 3-4-3), "RCS Pressure and Temperature (P/T) Limits," and
2. Specification 3/4-4-9 (ITS 3-4-12), "Overpressure Protection System."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. The analytical methods used to determine the RCS pressure and temperature limits were developed in accordance with:

10CFR 50 Appendix G and H
 Regulatory Guide 1-99, Revision 2
 NUREG-0800, Standard Review Plan Section 5.3.2
 Branch Technical Position MTEB 5-2
 ASME B&PV Code Section III, Appendix G
 ASME B&PV Code, Section XI, Appendix A
 WCAP-14040-NP-A, Section 2.2

2. Low Temperature Overpressure Protection limits (PORV pressure relief setpoint and LTOP enable temperature) were developed in accordance with:

NUREG-0800, Standard Review Plan Section 5.2.2
 Branch Technical Position RSB 5-2
 10CFR 50 Appendix G and H
 Regulatory Guide 1-99, Revision 2
 Branch Technical Position MTEB 5-2
 WCAP-14040-NP-A, Section 2.2

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

PAM Report

When a report is required by Specification 3/4 3.6, a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

ADMINISTRATIVE CONTROLS

03-09-LG

RECORD RETENTION (Continued)

- ~~d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications;~~
- ~~e. Records of changes made to procedures required by Specification 6.8.1;~~
- ~~f. Records of radioactive shipments;~~
- ~~g. Records of sealed source and fission detector leak tests and results; and~~
- ~~h. Records of annual physical inventory of all sealed source material of record.~~

~~6.10.2 The following records shall be retained for the duration of the unit Operating License:~~

- ~~a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report;~~
- ~~b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories;~~
- ~~c. Records of radiation exposure for all individuals entering radiation control areas;~~
- ~~d. Records of gaseous and liquid radioactive material released to the environs;~~
- ~~e. Records of transient or operational cycles for those unit components identified in Table 5.7-1;~~
- ~~f. Records of reactor tests and experiments;~~
- ~~g. Records of training and qualification for current members of the unit staff;~~
- ~~h. Records of in service inspections performed pursuant to these Technical Specifications;~~
- ~~i. Records of Quality Assurance activities required by the QA Manual;~~
- ~~j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10CFR 50.59;~~
- ~~k. Records of meetings of the PSRC and NSOC;~~

ADMINISTRATIVE CONTROLS

RECORD RETENTION (Continued)

03-09-LG

- ~~l. Records of analyses required by the Radiological Environmental Monitoring Program;~~
- ~~m. Records of the service lives of all hydraulic and mechanical snubbers required by the Final Safety Analysis Report including the date at which the service life commences and associated installation and maintenance records; and~~
- ~~n. Records of secondary water sampling and water quality.~~
- ~~o. Records of reviews performed for changes made to the RADIOLOGICAL MONITORING AND CONTROLS PROGRAM, OFFSITE DOSE CALCULATION PROCEDURE, ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURE, and the PROCESS CONTROL PROGRAM.~~

6.11 RADIATION PROTECTION PROGRAM

03-10-LG

~~Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.~~

6.12 HIGH RADIATION AREA

~~6.12.1 Pursuant to paragraph 20.203(c)(5) 20.1601⁹ of 10CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) requirements of 10CFR 20.1601, each high radiation area, as defined in 10CFR Part 20, in which the intensity of radiation is greater than 100 mrem/hr but equal to or less than 1000 mR/h at 30 cm at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of work permits for radiation (WPR). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the WPR issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h at 30 cm, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:~~

03-11-A

- ~~a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or~~
- ~~b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or~~

ADMINISTRATIVE CONTROLS

HIGH RADIATION AREA (Continued)

- c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Radiation Protection Manager in the WPR.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than or equal to 1000 mR/h at 45 cm (18 in.) at 30 cm from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors or continuously guarded to prevent unauthorized inadvertent entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved WPR which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the WPR, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

03-11-A

03-20-LS3

03-19-A

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h at 30 cm that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

03-11-A

03-20-LS3

6.13 PROCESS CONTROL PROGRAM (PCP)

03-12-LG

~~6.13.1 The PCP shall be approved by the Commission prior to implementation.~~

~~6.13.2 Changes to the PCP:~~

- ~~a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.o. This documentation shall contain:~~
- ~~1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and~~
 - ~~2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.~~
- ~~b. Shall become effective after review and acceptance by the PSRC and the approval of the Plant Manager.~~

ADMINISTRATIVE CONTROLS

moved to program

6.14 ~~RADIOLOGICAL MONITORING AND CONTROLS PROGRAM (RMCP), OFFSITE DOSE CALCULATION PROCEDURE (ODCP) and ENVIRONMENTAL RADIOLOGICAL MONITORING PROCEDURE (ERMP) OFFSITE DOSE CALCULATION MANUAL (ODCM)~~

6.14.1 ~~The RMCP, ODCP and ERMP shall be approved by the Commission prior to implementation.~~

a. ~~The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and~~

02-09-A

02-09-A

b. ~~The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specification 6.9.1.5 and Specification 6.9.1.6~~

6.14.2 ~~Changes to the RMCP, ODCP, and ERMP ODCM:~~

a. ~~Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.e. This documentation shall contain:~~

03-09-LG

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the level of radioactive effluent control required by 10CFR 20.10620.1302, 40 CFR Part 190, 10CFR 50.36a, and Appendix I to 10CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

02-09-A

b. ~~Shall become effective after review and acceptance by the PSRC and the approval of the Plant Manager.~~

02-13-LG

c. ~~Shall be submitted to the Commission in the form of a complete, legible copy of the entire RMCP, ODCP and ERMP ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the RMCP, ODCP or ERMP ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.~~

03-06-A

02-09-A

Methodology For Mark-Up of Current TS

This Enclosure contains the electronic (or hand written) mark-up of the current Technical Specifications (TS). The electronic (or hand written) mark-up is performed in accordance with the following guidelines:

- The current specifications are marked-up to reflect what they would look like when the substance of NUREG-1431 Revision 1 is incorporated.
- In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.
- Changes are identified by a change number in the right margin. A description/justification for each change is contained in Enclosure 3A.

There are four types of changes:

1. Deletions - Material is no longer in the specifications. (This includes material which is moved to the Bases of the TS.)
2. Additions - This includes the addition of new requirements, restrictions, etc. to the specifications which are not in the current TS.
3. Modifications - This includes requirements which exist in the current TS but are being revised in the improved TS.
4. Administrative - These are non-technical changes to the TS. These include adopting the new format of the improved STS, moving the location of material within the specifications, etc.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletion is identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number in the adjacent right margin.
- Modifications - The information being revised is annotated in the current TS using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number in the adjacent right margin.
- Administrative - The text of the current TS is not modified to reflect administrative changes. Where the administrative change might cause confusion to a reviewer, the change is identified by a change number in the right margin. For example, if a requirement is relocated to a specification in the improved TS which does not correspond with the specification in which that requirement is located in the current TS, a change number is provided in the mark-up of the current TS and an explanation is provided in Enclosure 3A which explains where that requirement has been located in the improved TS.

Methodology For Mark-Up of Current TS (Continued)

CHANGE NUMBERS:

A change number, located in the right margin adjacent to a technical change mark-up, provides an identifier for its corresponding description/justification and indicates the type of NSHC used. The change number is of the form 4-13-LS. The first number (i.e., 4 in this example) is a number assigned to each LCO (or group of similar LCOs) such that it refers to the same specification for each member utility in the Joint Licensing Subcommittee (JLS) regardless of the actual TS number in their individual Technical Specifications. A table of the change number prefixes versus each plant's specification numbers is provided in Enclosure 3A. The next set of numbers (i.e., -13 in this example) is an assigned number to identify changes within a given specification (i.e., having the same prefix number). As a result of differences between the individual JLS member current specifications and because of changes that may occur after initial number assignments, the numbers may not appear sequentially in the TS mark-up. The letter suffix (i.e., LS in this example) indicates the type NSHC used (e.g., A, M, LG, TR, LS, R).

In summary, changes may be annotated electronically or by using a hand mark-up. For electronic mark-up, "red-line" is used to annotate new information, "strike-out" is used to annotate deleted material (which includes material that is moved out of the specifications), and change numbers are used in the right margin to identify technical changes. All technical changes (i.e., "red-line" or "strike-out" items) require a change number. In addition, certain administrative changes (e.g., requirements moved to another specification) are also assigned a change number to provide additional clarification.

ENCLOSURE 3A

DESCRIPTION OF CHANGES TO CURRENT TS

Technical Specification Conversion Change Numbers	(1 Page)
Description of Changes	(7 Pages)

TECHNICAL SPECIFICATION CONVERSION CHANGE NUMBERS

SECTION 3/4.1

TECHNICAL SPECIFICATION TITLE	CHG NO.	CALLAWAY	WOLF CREEK	COMANCHE PEAK	DIABLO CANYON
Responsibility	01	6.1	6.1	6.1	6.1
Organization	01	6.2	6.2	6.2	6.2
Unit Staff Qualifications	01	6.3	6.3	6.3	none
Plant Staff Qualifications	01	none	none	none	6.3
Training	01	6.4	6.4	6.4	6.4
Review and Audit	01	DELETED	6.5	6.5	DELETED
Reportable Event Action	02	6.6	6.6	6.6	6.6
Safety Limit Violation	02	6.7	6.7	6.7	6.7
Procedures and Programs	02	6.8	6.8	6.8	6.8
Reporting Requirements	03	6.9	6.9	6.9	6.9
Record Retention	03	6.10	6.10	none	6.10
Radiation Protection Program	03	6.11	6.11	6.11	6.11
High Radiation Area [Optional]	03	6.12	6.12	6.12	6.12
Process Control Program (PCP)	03	6.13	6.13	6.13	6.13
Offsite Dose Calculation Manual (ODCM)	02	6.14	6.14	6.14	none
Radiological Monitoring and Controls Program (RMCP), Offsite Dose Calculation Procedure (ODCP) and Environmental Radiological Monitoring Procedure (ERMP)	02	none	none	none	6.14

DESCRIPTION OF CHANGES TO TS SECTION 6.0

This Enclosure contains a brief description/justification for each marked-up change to existing current plant Technical Specifications (CTS). The changes are keyed to those identified in Enclosure 2 (mark-up of the CTS). The referenced No Significant Hazards Considerations (NSHC) are contained in Enclosure 4. All proposed technical changes to the CTS are discussed below; however, some administrative changes (i.e., format, presentation, and editorial changes made to conform to the Improved Technical Specifications (ITS)) may not be discussed. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is specific and is not common to all the Joint Licensing Subcommittee (JLS) Plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-01	A	The "Responsibility" section is revised to be consistent with NUREG-1431 and current plant practice. The requirement to issue a management directive annually (i.e., control room command function) is deleted. The TS already adequately defines the function, and therefore the management directive is redundant.
01-02	A	The "Plant/Unit Staff" section is revised consistent with NUREG-1431 and plant practice. Sections are revised to reflect the shift crew composition table removal (if applicable), nonlicensed personnel, and changes made to the section to be on a unit basis versus plant basis consistent with NUREG-1431. Various editorial changes are made to accomplish the removal of the table and revisions to be consistent with NUREG-1431 and current plant practice.
01-03	A	The requirement for a senior reactor operator (SRO) to be present during fuel handling and to supervise all core alterations is not retained in ITS. This requirement is deleted consistent with NUREG-1431. This requirement essentially duplicates the regulation in 10CFR 50.54(m)(2)(iv). Since the requirements are not revised, this change is administrative only.
01-04	LG	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-05	A	The requirement for the presence of a reactor operator (RO) or an SRO in the control room is deleted from the TS since the requirement is consistent with and duplicative of the manning requirement in 10CFR 50.54(m)(2)(iii). Deletion of the CTS requirements does not change the manning requirements, and is therefore considered an administrative change.
01-06	LG	The details regarding the minimum shift crew requirements have been removed from the CTS because they are redundant to 10CFR 50.54(k), (l), and (m), with the exception of the requirement for nonlicensed operators. The corresponding ITS Section 5.2.2b requires meeting the requirements of these regulations which specify the shift complement regarding licensed operators for all modes of operation. The minimum shift crew requirements will be moved to a licensee controlled document.

DESCRIPTION OF CHANGES TO TS SECTION 6.0
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
01-07	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-08	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-09		Not Used.
01-10	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-11	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-12	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-13	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-14	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-01	A	CTS Section [6.6a.] for REPORTABLE EVENT ACTIONS has been deleted from the CTS. This section only repeats the regulatory reporting requirements defined in 10CFR 50.72 and 10CFR 50.73, and is unnecessary in the TS. Deletion of this section from CTS does not impact safety because it is redundant to the regulations cited, and is therefore acceptable.
02-02	LS-4	CTS Section [6.7], "Safety Limit Violation," requirements to notify the NRC within one hour following a violation of a safety limit (SL), submit a Safety Limit Violation Report and not resume plant operation until authorized by the Commission are being deleted. These requirements are a duplication of 10CFR 50.36(c)(1), 10CFR 50.72 and 10CFR 50.73. [The 14-day Safety Limit Violation Report in the CTS is not required since 10CFR 50.73 would require a 30 day Licensee Event Report.] Since the plant must meet the applicable requirements contained in the regulations, sufficient regulatory controls are maintained to allow removing these duplicate regulatory requirements from the CTS. The notification requirement to management and the review committees is an after-the-fact notification and is not necessary to assure safe operation of the facility. As such, this requirement is not necessary to be included in the TS. These changes are consistent with NUREG-1431 and traveler TSTF-5.

DESCRIPTION OF CHANGES TO TS SECTION 6.0
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
02-03	A	The implementation procedure requirements related to [] the PROCESS CONTROLPROGRAMS and the radiological environmental and OFFSITE DOSE CALCULATION PROGRAMS are deleted from the CTS consistent with NUREG-1431. These types of procedures are either required by Regulatory Guide (RG) 1.33, Rev. 2, Feb.1978 (referenced in the ITS or CTS), covered under the provisions of ITS 5.4.1.e, or required by 10CFR 50, Appendix E, and 10CFR 50.54(p) and (q). Therefore, these requirements are duplicative and unnecessary.
02-04	LG	The In-Plant Radiation Monitoring Program is based on NUREG-0737, Item III.D.3.3, "Improved Inplant Iodine Instrumentation Under Accident Conditions." The CTS requirement for this program is deleted but the program commitment will be in the Final Safety Analysis Report (FSAR) Update and implementing procedures. Any changes to the programmatic provisions in the FSAR Update would be controlled by 10CFR 50.59. Moving this information maintains consistency with NUREG-1431, Rev. 1.
02-05	A	Revises the Radioactive Effluent Controls Program to reflect wording in NUREG-1431 and format of the Administrative Controls section. The term "OPERABILITY" is replaced with "functional capability" to avoid confusion with the TS definition of OPERABILITY.
02-06	A	Consistent with NUREG-1431, the ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM is deleted. The details and description of the program are duplicative of [Offsite Dose Calculation Manual (ODCM)] requirements. The program only maintains consistency with the requirements of 10CFR 50, Appendix I. The [ODCM] and regulatory requirements provide sufficient control of these provisions, and therefore removing them from the CTS is acceptable.
02-07	A	Revises Radioactive Effluent Controls Program dose projections to meet original intent of TS prior to implementation of Generic Letter (GL) 89-01. GL 89-01 provided the wording for the STS (Section 5.5.4.e) which combined the requirements for cumulative and projected dose. This requires a plant to make projected doses for the quarter and year on a 31 day basis. It is only necessary and reasonable to make a projection for the next 31 days. A cumulative dose projection is still required for the current calendar quarter and year in accordance with the ODCM. This change is consistent with traveler WOG-72.
02-08	M	Revises the procedures section to refer to all programs listed in the program section of the Administrative Controls and ensures implementing procedure control for these programs. This change is consistent with NUREG-1431. This change is more restrictive because it now includes all programs (current and added) in the TS as well as those already specified in this paragraph.

DESCRIPTION OF CHANGES TO TS SECTION 6.0
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
02-09	A	The description of the [ODCM] (or equivalent programs and procedures) was revised to be consistent with NUREG-1431. The [ODCM] description is also revised to reflect new 10CFR Part 20 requirements.
02-10	M	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-11	M	New program requirements, "Safety Function Determination Program" and "Bases Control Program," would be added consistent with NUREG-1431. Although these new programs reflect current plant practice, delineating them in the ITS would be more restrictive.
02-12	LG	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-13	LG	Revises Section 6.14, item b, to move the requirement that ODCM (or similar programs and procedures) changes require review and acceptance by onsite review committees to the ODCM. The onsite review of ODCM changes is currently required per [procedures]. This change is consistent with NUREG-1431.
02-14	M	Per GL 89-01, concentrations of radioactive material releases in liquid effluents to unrestricted area shall conform to 10 times the concentration values in Appendix B, Table 2, Column 2, of 10CFR 20.1001-20.2401. Proposed traveler pending.
02-15	LG	CTS Section [6.6b.] contains requirements for the plant review and submittal of a REPORTABLE EVENT. This information is to be moved to a licensee controlled document. Given that these reviews and submittal of results are required following the event without a specified Completion Time, the requirements are not necessary. The moving of this information maintains consistency with NUREG-1431, Rev. 1.
02-16	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-17	LS-1	The "Reactor Coolant Pump Flywheel" section is being revised consistent with Westinghouse Owners Group (WOG)-85. The proposed changes provide an exception to the examination requirements in RG 1.14, Rev. 1, "Reactor Coolant Pump Flywheel Integrity." The proposed exception to the recommendations of Regulatory Position C.4.b would allow for an acceptable inspection method of either an ultrasonic volumetric or surface examination. The acceptable inspection method would be conducted at approximately 10 year intervals. This change is consistent with the NRC Safety Evaluation Report (SER) associated with WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination."

DESCRIPTION OF CHANGES TO TS SECTION 6.0
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
02-18	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-19	LS-2	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
02-20	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).
03-01	A	Revises "Routine Reports" section to be consistent with NUREG-1431. The method for submitting all reports is revised to be in accordance with 10CFR 50.4. Since this change merely makes the TS consistent with the regulations, it is considered administrative.
03-02	A	The requirement to submit a startup report is deleted from the CTS to be consistent with NUREG-1431. This report required no staff approval and was submitted after the fact, and therefore is not required to ensure safe plant operation. The approved 10CFR 50, Appendix B, Quality Assurance (QA) Plan, and FSAR Startup Testing Program provides assurance that the affected activities are adequately performed and that appropriate corrective actions, if required, are taken.
03-03	A	The Annual Reports section is revised to be consistent with NUREG-1431 and traveler TSTF-152. Names and formats are revised consistent with NUREG-1431. Also, this change revises the Annual Report section to reflect the new 10CFR Part 20 requirements and associated recommended changes noted in NRC letter dated July 28, 1995, "Changes to Technical Specifications Resulting from 10CFR 20 and 50.36a Changes." (From Christopher I. Grimes to Owners Groups Chairs)
03-04	A	The requirement to report specific activity limit violations is deleted consistent with NUREG-1431. This report is a history of reactor coolant system (RCS) specific activity limiting condition for operation (LCO) entries. GL 83-43 and revised reporting requirements in the regulations intended that LCO entry reports no longer be required. The reporting requirements in regulations cover situations such as seriously degraded barriers (fuel failure). Therefore, every violation of the RCS specific activity LCO need not be reported. Serious degradation of a fission product barrier, among other more serious events are required to be reported by 10CFR 50.73. This change is administrative in that it only affects reports and does not affect plant operations.
03-05	A	Not applicable to DCP. See Conversion Comparison Table (Enclosure 3B).

DESCRIPTION OF CHANGES TO TS SECTION 6.0
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
03-06	A	CTS [6.9.1.6], "Annual Radioactive Effluent Release Report" and CTS [6.14c.] are revised consistent with NUREG-1431, Rev. 1, to delete the term "Annual" and modify the submittal date. This change provides a reference to 10CFR 50.36a since 10CFR specifies that the report must be submitted annually and include the results from the previous 12 months of operation.
03-07	A	CTS [6.9.1.5], "Annual Radiological Environmental Operating Report" is revised to include specific details concerning the contents of the report. This change is consistent with NUREG-1431, Rev. 1.
03-08	A	CTS Specifications [6.9.1.7, 6.9.1.8 and 6.9.2] are revised to delete the reference to submittal location for the monthly report, CORE OPERATING LIMITS REPORT (COLR), and special reports. The requirements related to report submittal are contained in 10CFR. Since conformance to 10CFR is a condition of the license, specific identification of this requirement in the TS would be duplicative and is not necessary. Since the plant requirements remain the same, the change is considered an administrative change. This change is consistent with NUREG-1431, Rev. 1.
03-09	LG	The record retention requirements are moved to the FSAR and implementing procedures. The removal of this detail from the CTS is consistent with NUREG-1431. The requirement for retention of records related to activities affecting quality is contained in 10CFR 50, Appendix B, Criteria XVII and other sections of 10CFR 50 that are applicable to the plant (i.e., 50.71, etc.). Post-completion review of records does not directly assure operation of the facility in a safe manner, as the activities described in the documents have already been performed. By retaining these requirements in plant procedures and licensee controlled documents, any changes in these record retention requirements will be adequately controlled under the provisions of 10CFR 50.59 and the applicable regulations.
03-10	LG	The Radiation Protection Program is moved to the FSAR consistent with NUREG-1431. This program requires procedures to be prepared for personnel radiation protection consistent with 10CFR Part 20. These procedures are for the protection of nuclear plant personnel and have no impact on nuclear safety or the health and safety of the public. Requirements to have procedures to implement 10CFR Part 20 are contained in 10CFR 20.1101(b). Periodic review of these procedures is required by 10CFR 20.1101(c). The CTS is redundant to requirements in the regulations and thus is deleted.
03-11	A	The high radiation area is revised to be consistent with NUREG-1431 and the new Part 20 requirements. Changes are nontechnical to add clarification and conform with NUREG-1431 and RG 8.38.

DESCRIPTION OF CHANGES TO TS SECTION 6.0
(Continued)

<u>CHANGE NUMBER</u>	<u>NSHC</u>	<u>DESCRIPTION</u>
03-12	LG	The PROCESS CONTROL PROGRAM (PCP) section is proposed to be moved outside the CTS consistent with NUREG-1431. The PCP implements the requirements of 10CFR 20, 10CFR 61, and 10CFR 71. Therefore, relocation of the description of the PCP from the CTS does not affect the safe operation of the facility. The PCP will be adequately described in licensee controlled documents.
03-13	M	The following report[s] will be added to the ITS Administrative Controls section: "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" [and "Post Accident Monitoring (PAM) Report"]. The PTLR is more restrictive because it is not currently required. This change is also described in the description of changes for CTS Section 3/4.4 (PTLR). [The PAM Report is already required per TS 3/4.3; however, delineating the report requirement in the ITS is considered to be more restrictive.]
03-14	M	SHUTDOWN MARGIN (SDM) values would be moved to the COLR traveler TSTF-9. In addition, moderator temperature coefficient (MTC) limits would also be moved to the COLR. The addition of these specifications to the COLR is considered to be more restrictive.
03-15	M	Refueling boron concentration limits will also be added to the COLR. The addition of these limits to the COLR is considered to be more restrictive.
03-16	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
03-17	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
03-18	A	Not applicable to DCCP. See Conversion Comparison Table (Enclosure 3B).
03-19	A	The term "unauthorized" is changed to "inadvertent" in the High Radiation Area section. The prevention of inadvertent entry is discussed in Section 1.5 of RG 8.38. This RG reflects the NRC's position regarding physical barriers for high radiation areas. Radiation areas within the limits listed shall be locked or continuously guarded to prevent inadvertent entry as discussed in RG 8.38. Furthermore, the distinction between unauthorized versus inadvertent is important based on a notice of violation that Callaway received on this interpretation of terms.
03-20	LS-3	Consistent with NUREG-1431, the use of a continuous guard is provided as an additional option for preventing inadvertent entry into high radiation areas that are accessible to individuals. This option is acceptable because it provides access control to these areas commensurate with the other methods already permitted by the CTS (i.e., locked doors, or barricades with posted signs and flashing lights for isolated areas without enclosures).

ENCLOSURE 3B

CONVERSION COMPARISON TABLE - CURRENT TS

Conversion Comparison Table

(8 pages)

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	The "Responsibility" section is revised to delete the requirement to issue a management directive annually (i.e., control room command function). The TS already adequately defines the function, and therefore the management directive is redundant.	Yes	Yes	Yes	Yes
01-02 A	The "Plant/Unit Staff" section is revised to reflect the shift crew composition table removal (if applicable), nonlicensed personnel, and changes made to the section to be on a unit basis versus plant basis. Various editorial changes are made to accomplish the removal of the table and revisions to be consistent with NUREG-1431 and current plant practice.	Yes	No, CTS already incorporates changes.	Yes	Yes
01-03 A	The requirement for an SRO to be present during fuel handling and to supervise all CORE ALTERATIONS is not retained in ITS. This requirement essentially duplicates the regulation in 10CFR 50.54(m)(2)(iv).	Yes	No, deleted per CTS Amendment 50/36.	Yes	Yes
01-04 LG	The details of the review and audit, independent safety engineering group, and training functions are removed from the CTS. Those items not specifically covered by a regulation are moved to licensee controlled documents; otherwise the requirements are deleted.	No, deleted per CTS Amendment 117/115.	No, deleted per CTS Amendment 50/36.	Yes, move to USAR.	Yes, move to FSAR and OQAM.
01-05 A	The requirement for the presence of an RO or an SRO in the control room is deleted from the TS since the requirement is adequately controlled by 10CFR 50.54(m)(2)(iii).	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-06 LG	The details regarding the minimum shift crew requirements have been removed from the CTS because they are redundant to 10CFR 50.54(k), (l), and (m) with the exception of the requirement for nonlicensed operators. The minimum shift crew requirements will be moved to a licensee controlled document.	Yes, move to FSAR.	No, CTS already contains changes.	Yes, move to USAR.	Yes, move to FSAR.
01-07 LG	Revises Section 6.2.2a, Unit Staff Organization, to reflect the nonlicensed operator staffing requirements for a single unit site. The minimum shift crew composition as described in Table 6.2-1 has been moved to a licensee controlled document.	No, DCPD is a multi-unit plant.	No, CPSES is a multi-unit plant.	Yes, moved to USAR Chapter 13.	Yes, moved to FSAR.
01-08 LG	Moves the fire brigade requirements to a licensee controlled document. These requirements can be found in BTP ASB 9.5-1 and their duplication on the ITS is not required.	No, LA 75/74.	Yes, move to FSAR.	Yes, move to USAR.	Yes, move to FSAR.
01-09	Not Used.	NA	NA	NA	NA
01-10 M	Adds requirement for three auxiliary operators for the two unit sites with both units shut down or defueled.	No, already DCPD requirement.	Yes	No, Wolf Creek is a single unit plant.	No, Callaway is a single unit plant.
01-11 A	For clarity, a note is added to state that one radiation protection technician and one chemistry technician can fulfill the staffing requirements for both units.	No, DCPD procedure and operational requirements differ.	Yes	No, Wolf Creek is a single unit site.	No, Callaway is a single unit site.
01-12 A	Deletes the Comanche Peak STA qualifications based on use of RG 1.8, Rev. 2.	No	Yes	No	No
01-13 A	Adds new statement to accommodate unexpected absences of on-duty crew member.	No, already in CTS.	Yes	No, already in CTS.	No, already in CTS.

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-14 A	Deletes the shift supervisors and operating supervisor from Section 6.2 as required to hold a SRO license.	No, DCPD procedure and operational requirements differ.	No, not in CTS.	No, Wolf Creek has different requirements.	Yes
02-01 A	CTS Section [6.6a.] for REPORTABLE EVENT actions has been deleted from the CTS.	Yes	No, deleted from CTS per Amendment 50/36.	Yes	Yes
02-02 LS-4	The "Safety Limit Violation" section is deleted from the CTS. The NRC reporting requirements are duplicative of 10CFR 50.72, 10CFR 50.73, and 10CFR 50.36.	Yes	Yes	Yes	Yes
02-03 A	The implementation procedure requirements related to [] the PCPs and the Radiological Environmental and Offsite Dose Calculation Programs are deleted from the CTS.	Yes	No, deleted from CTS per Amendment 50/36.	Yes	Yes
02-04 LG	The In-Plant Radiation Monitoring Program is deleted from the CTS but the program commitment will be in the FSAR and implementing procedures. Any changes to the programmatic provisions in the FSAR would be controlled by 10CFR 50.59.	Yes, moved to FSAR.	No, deleted from CTS per Amendment 50/36.	Yes, moved to USAR.	Yes, moved to FSAR.
02-05 A	Revises the radioactive effluent controls program to reflect wording in NUREG-1431 and format of the Administrative Controls section. The term "OPERABILITY" is replaced with "functional capability" to avoid confusion with the TS definition of OPERABILITY.	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-06 A	Radiological Environmental Monitoring Program is deleted. The details and description of the program are duplicative of [ODCM] requirements. The program only maintains consistency with the requirements of 10CFR 50, Appendix I.	Yes	No, deleted from CTS per Amendment 50/36.	Yes	Yes
02-07 A	Revises Radioactive Effluent Controls Program dose projections to meet original intent of TS prior to implementation of GL 89-01. (WOG-72)	Yes	Yes	Yes	Yes
02-08 M	Revises the procedures section to refer to all programs listed in the program section of the Administrative Controls and ensures implementing procedure control for these programs.	Yes	Yes	Yes	Yes
02-09 A	The description of the [ODCM] (or equivalent programs and procedures) is revised. The [ODCM] description is also revised to reflect new 10CFR Part 20 requirements.	Yes	No, changes included in CTS.	Yes	Yes
02-10 M	Adds new program called "Technical Requirements Manual (TRM)."	No, DCPD does not have TRM program.	Yes	No, Wolf Creek does not have a TRM program.	No, Callaway does not have a TRM program.
02-11 M	New program requirements, "Safety Function Determination Program" and "Bases Control Program" would be added.	Yes	Yes	Yes	Yes
02-12 LG	This change moves the Emergency Diesel Generator Reliability Program requirement to a licensee controlled document.	No, DG failure reports addressed in Section 3/4.8.	No, not in CTS.	Yes, moved to USAR.	Yes, moved to FSAR.
02-13 LG	Revises Section 6.14 item b to move the requirement that ODCM (or similar programs and procedures) changes require review and acceptance by onsite review committees to the ODCM.	Yes, move to ODCM.	Yes, moved to ODCM.	Yes, requirements are in AP 07B-003 (ODCM).	Yes, ODCM.

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-14 M	Per GL 89-01, concentrations of radioactive material releases in liquid effluents to unrestricted area shall conform to 10 times the concentration values in Appendix B, Table 2, Column 2 of 10CFR 20.1001-20.2401.	Yes	No, changes included in CTS.	Yes	Yes
02-15 LG	CTS Section [6.6b.] contains requirements for the plant review and submittal of a REPORTABLE EVENT. This information is to be moved to a licensee controlled document.	Yes	No, deleted from CTS per Amendment 50/36.	Yes	Yes
02-16 A	Change the Diesel Fuel Oil Testing Program description for sampled properties of new fuel oil from "within limits" to "analyzed" within 30 days following sampling and addition of the fuel oil to storage tanks. This wording more clearly defines that within 30 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D 975-81 are met. This change is consistent with the Bases for SR 3.8.3.3.	No, not in CTS.	No, not in CTS.	Yes	Yes
02-17 LS-1	"Reactor Coolant Pump Flywheel" is being revised consistent with WOG-85. The proposed changes provide an exception to the examination requirements in RG 1.14, Rev. 1, "Reactor Coolant Pump Flywheel Integrity."	Yes	No, see Section 3/4.4, CN 10-03-LS.	No, LAR submitted 12/3/96.	Yes
02-18 A	Revise the Radiological Effluent Controls Program dose rate limits to reflect changes to 10CFR Part 20, draft GL, and proposed traveler.	No, already in CTS.	No, already in CTS.	Yes	Yes
02-19 LS-2	The surveillance interval for verifying that other properties are within limits for ASTM 2D fuel oil is changed from "within 30 days" to within 31 days" after obtaining a sample.	No, addressed in 3/4.8 (CN 01-60-LS24).	No, addressed in 3/4.8 (CN 01-60-LS24).	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-20 A	Add the provisions of Specifications 3.0.2 and 3.0.3 are applicable to the Diesel Fuel Oil Testing Program. This change is consistent with TSTF-118.	No, not in CTS.	No, not in CTS.	Yes	Yes
03-01 A	The method for submitting all reports is revised to be in accordance with 10CFR 50.4.	Yes	Yes	Yes	Yes
03-02 A	The requirement to submit a startup report is deleted from the CTS This report required no staff approval and was submitted after the fact, and is therefore not required to ensure safe plant operation. The approved 10CFR 50, Appendix B, QA Plan, and FSAR Startup Testing Program provides assurance that the affected activities are adequately performed and that appropriate corrective actions, if required, are taken.	Yes	No, deleted from CTS per Amendment 50/36.	Yes	Yes
03-03 A	Revises the annual report section to reflect the new 10CFR Part 20 requirements and associated recommended changes noted in NRC letter dated July 28, 1995, "Changes to Technical Specifications Resulting from 10CFR 20 and 50.36a Changes." (From Christopher I. Grimes to Owners Groups Chairs) - TSTF-152	Yes	Yes, except the Part 20 requirements were removed from the TS in Amendment 50/36.	Yes	Yes
03-04 A	The requirement to report specific activity limit violations is deleted consistent with NUREG-1431. Serious degradation of a fission product barrier, among other more serious events, are required to be reported by 10CFR 50.73. This change is administrative in that it only affects reports and does not affect plant operations.	Yes	Yes	Yes	Yes
03-05 A	The Annual Radiological Environmental Operating Report, including the submittal date, is revised.	No, DCPD report dates to remain as in CTS.	Yes	No, WCNOG report dates to remain as in CTS.	No, Callaway report dates to remain as in CTS.

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-06 A	CTS [6.9.1.6], "Annual Radioactive Effluent Release Report," and CTS [6.14c.] are revised consistent with NUREG-1431, Rev. 1, to delete the term "Annual" and modify the submittal date.	Yes	Yes	Yes	Yes
03-07 A	CTS [6.9.1.5], "Annual Radiological Environmental Operating Report," is revised to include specific details concerning the contents of the report.	Yes	Yes	Yes	Yes
03-08 A	CTS Specification [6.9.1.7, 6.9.1.8 and 6.9.2] are revised to delete the reference to submittal location for the monthly report, CORE OPERATING LIMITS REPORT, and special reports.	Yes	Yes	Yes	Yes
03-09 LG	The record retention requirements are moved to the FSAR and implementing procedures. The requirement for retention of records related to activities affecting quality is contained in 10CFR 50, Appendix B, Criteria XVII, and other sections of 10CFR 50 that are applicable to the plant (i.e., 50.71, etc.).	Yes	Yes	Yes	Yes
03-10 LG	The Radiation Protection Program is moved to the FSAR. This program requires procedures to be prepared for personnel radiation protection consistent with 10CFR Part 20. Periodic review of these procedures is required by 10CFR 20.1101(c).	Yes	No, deleted from CTS per Amendment 50/36.	Yes	Yes
03-11 A	The High Radiation Area section is revised to be consistent with the new Part 20 requirements. Changes are nontechnical to add clarification.	Yes	Yes	Yes	Yes
03-12 LG	The PCP section is proposed to be moved outside the CTS. The PCP implements the requirements of 10CFR 20, 10CFR 61, and 10CFR 71.	Yes, move to FSAR.	No, deleted from CTS per Amendment 50/36.	Yes, move to USAR.	Yes, move to FSAR.

CONVERSION COMPARISON TABLE - CURRENT TS 6.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-13 M	The following report[s] will be added to the ITS Administrative Controls section: "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" [and "Post Accident Monitoring (PAM) Report"] .	Yes	Yes	Yes	Yes
03-14 M	SDM values would be moved to COLR per traveler TSTF-9. In addition, MTC limits would also be moved to the COLR.	Yes	No, already part of CTS.	Yes	Yes
03-15 M	Adds refueling boron concentration limits to COLR.	Yes	Yes	No, already in CTS.	Yes
03-16 A	Deletes one of the allowed ECCS evaluation models for CPSES Unit 2 which is no longer used.	No	Yes	No	No
03-17 A	Deletes the methodology section references in the COLR.	No, references do not exist in DCPD CTS.	Yes	Yes	Yes
03-18 A	Moves the reporting requirement for documentation of all challenges to the PORVs or safety valves to the WCNOG Monthly Operating Report.	No	No	Yes	No
03-19 A	The term "unauthorized" is changed to "inadvertent" in the High Radiation Area section. The prevention of inadvertent entry is discussed in Section 1.5 of RG 8.38.	Yes	Yes	Yes	Yes
03-20 LS3	The use of a continuous guard is provided as an additional option for preventing inadvertent entry into high radiation areas that are accessible to individuals.	Yes	Yes	Yes	No, maintaining CTS.

ENCLOSURE 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

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III. Generic NSHCs	
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"LG" - Less Restrictive (moving information out of the TS)	10
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IV. Specific NSHCs - "LS"	
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LS2	Not Applicable to DCPD
LS3	16
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I. NO SIGNIFICANT HAZARDS CONSIDERATIONS ORGANIZATION

In accordance with the provisions of 10CFR50.90, this License Amendment Request proposes to revise the CTS. The proposed revision includes converting the CTS to the Improved Standard Technical Specifications (ISTS) in NUREG-1431, Revision 1. The conversion to the ISTS (also referred to as the improved STS or ISTS) has generated a large number of changes. Evaluations pursuant to 10CFR50.92 showing that the proposed changes do not involve significant hazards considerations are provided for each TS chapter. However, due to the volume of changes, similar changes have been grouped in categories to facilitate the NSHCs required by 10CFR50.92.

Generic NSHCs have been developed that correspond to each category of changes. In addition, since each TS chapter has been evaluated individually, chapters may contain chapter-specific generic NSHCs. NSHCs for changes that cannot be grouped into a category have also been developed. Typically, less restrictive technical changes must be evaluated individually. Each TS chapter will, therefore, contain "change-specific" NSHCs for less restrictive technical changes as well as generic NSHCs.

Each change to the CTS is marked-up on the appropriate page and technical changes are assigned a change number. Obvious editorial or administrative changes are not marked-up. The change number in the right margin of the marked-up page is used in the Description of Changes (Enclosure 3A), which provides a detailed basis for each change and a reference to the applicable NSHC. For Enclosures 3A, 3B, 4, 6A, and 6B, text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

II. DESCRIPTION OF NSHC EVALUATIONS

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following are brief descriptions of the generic NSHCs contained within this TS chapter. The reference symbols are used in the Discussion of Changes to index the applicable NSHC for each change described and are incorporated into the change numbers. Additional generic subcategories may be developed and will be referenced by adding a numeric designator to the existing alpha reference symbol (i.e., LG1, LG2, A1, A2, etc).

Administrative

Reference symbol "A" (Administrative)

This category consists of changes which are editorial in nature, involve the movement of requirements within the TS without affecting their technical content, simply reformat a requirement, or clarify the TS (such as deleting a footnote no longer applicable due to a technical change to a requirement). It also includes nontechnical changes made to conform to the Writer's Guide or the ISTS in NUREG-1431. Most administrative changes have not been marked-up on the CTS, and thus are not specifically referenced to a discussion of change or NSHC. If no discussion of change or NSHC is referenced for a change it is considered administrative in nature and this generic NSHC applies. This NSHC may also be referenced in a discussion of change for an administrative change that is not obvious and requires an explanation.

Relocation of Technical Specification Requirements

Reference symbol "R" (Relocation)

This category applies to TS requirements that do not meet the criteria in 10CFR50.36(c)(2)(ii). TS requirements affected by the application of the criteria are annotated with an "R" in the description of the change (Enclosure 3A). The "R" designation and the description of the relocation direct the reviewer to this NSHC for a description and evaluation of the change.

Moving information out of Technical Specifications

Reference symbol "LG" (Less Restrictive, Generic)

In some cases, information will be moved out of the TS while the underlying requirement remains (e.g., the requirement for equipment operability is retained in the LCO but the definition of operability is moved to the Bases). The affected information maybe moved to the Bases, the Final Safety Analysis Report (FSAR), or other licensee controlled documents. This category of change is considered to be less restrictive (no longer controlled by TS) and usually involves moving information of a descriptive nature. These changes are generally made in order to conform with NUREG-1431 format and content.

Technical change, more restrictive

Reference symbol "M" (More Restrictive, Generic)

This category consists of changes that add new requirements to the TS or revise existing requirements to be more stringent. These changes are typically made to conform to applicable requirements of NUREG-1431.

II. DESCRIPTION OF NSHC EVALUATIONS

SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

Those TS changes that must be evaluated individually are typically the less restrictive technical changes. Each NSHC for less restrictive technical changes in this TS chapter will be numbered sequentially. The applicable NSHC for each less restrictive change will be referenced in the Description of Change (Enclosure 3A) for this chapter. The Description of Change contains the basis for the change.

Technical change, less restrictive

Reference symbol "LS" (Less Restrictive, Specific)

This category consists of changes which revise existing requirements such that more restoration time is provided, fewer compensatory measures are needed, or fewer or less restrictive surveillance requirements are required. This would also include requirements which are deleted from the TS (not relocated or moved to other documents).

Technical change, recurring - less restrictive

Reference symbol "TR-1, 2, 3...." (Technical Recurring)

This category consists of the same kind of changes as LS above except that they are generic to several specifications.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A" 10CFR50.92 EVALUATION FOR ADMINISTRATIVE REFORMATTING AND REWORDING

This proposed TS revision includes reformatting and rewording the remaining requirements in accordance with the NUMARC Technical Specification Writer's Guide and the Improved Standard Technical Specifications in NUREG-1431. This is intended to make the TS more readily understandable to plant operators and other users. Application of the Writer's Guide will also assure consistency between specifications. During this reformatting and rewording process, no technical changes (either actual or interpretational) were made to the TS unless they were identified and justified.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to the current Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, no question of safety is involved.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"A"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "A" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
10CFR50.92 EVALUATION
FOR
RELOCATING TECHNICAL SPECIFICATION REQUIREMENTS
TO OTHER LICENSEE CONTROLLED DOCUMENTS

This proposed TS revision includes relocating requirements, which do not meet the TS criteria, to documents with established control programs. Relocation of these requirements allows the TS to be reserved only for those conditions or limitations upon reactor operation which are necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety thereby focusing the scope of the TS.

Therefore, requirements which do not meet the TS criteria in 10CFR50.36(c)(2)(ii) have been relocated to other licensee controlled documents. This regulation addresses the scope and purpose of TS. In doing so, it sets forth a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the TS. These criterias are as follows:

- Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;
- Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;
- Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier; and
- Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

This proposed change has been evaluated and it is concluded that the change does not meet the criterias listed above. The Conversion Comparison Table (Enclosure 3B) specifies the proposed location of these relocated requirements.

TS requirements that do not meet the NRC's criteria are being relocated to other licensee controlled documents. Some of these requirements will be relocated to documents that are subject to the provisions of 10CFR50.59. This will ensure that changes to these relocated requirements will be limited to those that do not involve an unreviewed safety question. Other requirements will be relocated to other licensee documents which have similar regulatory controls (e.g., the Quality Assurance Plan, as described in the FSAR, which is controlled by 10CFR50.54a). The remainder of the requirements that do not meet the NRC criteria will be relocated to programs that are controlled via the Administrative Controls section of the improved TS. This will ensure an appropriate level of control over changes to these requirements. The TS change to relocate requirements has been reviewed by a multi-disciplinary group of responsible, technical supervisory personnel, including onsite operations personnel.

Compliance with the relocated requirements will not be affected by this proposed change to the current Technical Specifications. The required periodic surveillances will continue to be performed to ensure that limits on parameters are maintained. Therefore, relocation of these requirements will have no impact on system operability or the maintenance of controlled parameters within limits.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change relocates requirements and surveillances for structures, systems, components, or variables which did not meet the criteria for inclusion in the improved STS. The affected structures, systems, components, or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. These relocated operability requirements and surveillances will continue to be maintained pursuant to 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variables are the same as the current Technical Specifications. Since any future changes to these requirements and the associated surveillance procedures will be evaluated per the requirements of 10CFR50.59, other regulatory requirements (as applicable for the document to which the requirement is relocated), and/or the Administrative Controls section of the improved STS, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"R"
(Continued)

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "R" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
10CFR50.92 EVALUATION
FOR
MOVING INFORMATION FROM TECHNICAL SPECIFICATIONS TO TECHNICAL SPECIFICATION
BASES, FSAR OR OTHER LICENSEE
CONTROLLED DOCUMENTS

Some information that is descriptive in nature regarding the equipment, system(s), actions or surveillances identified by the specification has been removed from the proposed specification and included in the proposed Bases, FSAR, or other licensee controlled document. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner due to the controls which presently exist on the documents where the information is being moved.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change moves requirements from the TS to the Bases, FSAR, or other licensee controlled documents. The Bases, FSAR, or other licensee controlled documents containing the moved requirements will be maintained using the provisions of 10CFR50.59 or other appropriate controls.

Since any changes to the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to adequately limit the probability or consequences of an accident previously evaluated. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"LG"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the TS to the Bases, FSAR, or other licensee controlled documents are the same as the current TS. Since any future changes to these requirements in the Bases, FSAR, or other licensee controlled documents will be evaluated per the requirements of 10CFR50.59 or other appropriate regulatory controls, proper controls are in place to maintain an appropriate margin of safety. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LG" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
10CFR50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE MORE RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This proposed revision involves modifying the current Technical Specifications to impose more stringent requirements and achieves consistency with the proposed improved Standard Technical Specifications (NUREG-1431).

The current Technical Specifications have been modified in some areas to impose more stringent guidelines than previously required. These more restrictive modifications are being imposed to be consistent with the proposed improved Standard Technical Specifications (NUREG-1431). Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10CFR50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves NSHC, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change provides more stringent requirements for the improved TS. These more stringent requirements are not assumed initiators of analyzed events and will not alter assumptions relative to mitigation of accidents or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. GENERIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

"M"
(Continued)

3. Does this change involve a significant reduction in a margin of safety?

The imposition of more stringent requirements either has no impact on or increases the margin of plant safety by:

- a) Increasing the analytical or safety limit,
- b) Increasing the scope of the specification to include additional plant equipment or to add additional requirements,
- c) Increasing the applicability of the specification,
- d) Providing additional actions,
- e) Decreasing restoration times,
- f) Imposing new surveillances, or
- g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "M" resulting from the conversion to the improved TS format satisfy the NSHC standards of 10CFR50.92(c), and accordingly a no significant hazards consideration finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1

10CFR50.92 EVALUATION

FOR

TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with traveler WOG-85, the Reactor Coolant Pump Flywheel Inspection Program is revised to provide an exception to the examination requirements in RG 1.14, Rev 1. The exception (to Regulatory Position C.4.b(1) and C.4.b(2)) allows for an acceptable inspection method of either an ultrasonic volumetric or surface examination. The inspection would be conducted at 10-year intervals coinciding with the inservice inspection (ISI) schedule required by ASME Section XI. The acceptability of the proposed change is established in WCAP-14535, Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination, with Limitations. The NRC's safety evaluation of the topical concluded that the inspections should not be completely eliminated but should be conducted during scheduled ISI or reactor coolant pump (RCP) maintenance at approximately 10-year intervals. The proposed change is consistent with these recommendations.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The safety function of the RCP flywheels is to provide a coastdown period during which the RCPs would continue to provide reactor coolant flow to the reactor after loss of power to the RCPs. The maximum loading on the RCP flywheel results from overspeed following a loss-of-coolant accident (LOCA). The maximum obtainable speed in the event of a LOCA was predicted to be less than 1500 rpm. Therefore, a peak LOCA speed of 1500 rpm is used in the evaluation of RCP flywheel integrity in WCAP-14535. This integrity evaluation shows a very high flaw tolerance for the flywheels. The proposed change does not affect that evaluation. Reduced coastdown times due to a single failed flywheel is bounded by the locked rotor analysis; therefore, it would not place the plant in an unanalyzed condition. Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1 (continued)

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated since it will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced since the proposed change does not involve the addition or modification of equipment, nor does the change alter the design or operation of affected plant systems, structures, or components.

3. *Does this change involve a significant reduction in a margin of safety?*

The operating limits and functional capabilities of the affected systems, structures, and components are basically unchanged by the proposed change. Industry experience with flywheel inspections performed have identified no indications affecting flywheel integrity. As identified in WCAP-14535, detailed stress analysis as well as risk analysis have been completed with the results indicating that there would be no change in the probability of failure for RCP flywheels if all inspections were eliminated. Therefore, these changes do not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS1" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3

10 CFR 50.92 EVALUATION FOR RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Consistent with NUREG-1431, the use of a continuous guard is provided as an additional option for preventing inadvertent entry into high radiation areas that are accessible to individuals. This option is acceptable because it provides access control to these areas commensurate with the other methods already permitted by the CTS (i.e., locked doors, or barricades with posted signs and flashing lights for isolated areas without enclosures).

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

This change only provides an additional acceptable method for preventing inadvertent entry into high radiation areas that are accessible to individuals. The proposed change does not involve any new operating activities or hardware changes. Because the accident analyses are initiated from within the conditions defined by the TS LCO and these LCOs are unchanged, the accident analyses are unaffected. Thus, the change cannot affect any accident probability or consequences.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the proposed alternate method for preventing inadvertent entry into high radiation areas. Therefore, there is no possibility for a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3
(continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The accident analyses are assumed to be initiated from conditions which are consistent with the TS LCO. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no affect on any actual or regulated failure point which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-3" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4

10 CFR 50.92 EVALUATION FOR RECURRING TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

CTS Section [6.7], "Safety Limit Violation," requirements to notify the NRC within one hour following a violation of a SL, submit a Safety Limit Violation Report and not resume plant operation until authorized by the Commission are being deleted. These requirements are a duplication of 10 CFR 50.36(c)(1),

10 CFR 50.72, and 10 CFR 50.73. [The 14-day Safety Limit Violation Report in the CTS is not required since 10 CFR 50.73 would require a 30 day Licensee Event Report.] Since the plant must meet the applicable requirements contained in the regulations, sufficient regulatory controls are maintained to allow removing these duplicate regulatory requirements from the CTS. The notification requirement to management and the review committees is an after-the-fact notification and is not necessary to assure safe operation of the facility. As such, this requirement is not necessary to be included in the TS. These changes are consistent with NUREG-1431 and traveler TSTF-5.

This proposed TS change has been evaluated and it has been determined that it involves NSHC. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92© as quoted below:

"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

These changes are only administrative in nature. The proposed change does not involve any new plant operating activities or hardware changes. The accident analyses are unaffected. Thus, the change cannot affect any accident probability or consequences.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

The assumptions of the accident analyses are unaffected by the proposed change. No new permutations or event initiators are introduced by the administrative changes. Therefore, there is no possibility for a new or different kind of accident.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS4 (continued)

3. *Does this change involve a significant reduction in a margin of safety?*

The accident analyses are assumed to be initiated from conditions which are consistent with the TS LCO. The proposed change does not affect any LCO. Therefore, there is no change in the accident analyses and all relevant event acceptance criteria remain valid. Further, the proposed change has no affect on any actual or regulated failure point, which is protected by an event acceptance criterion. Because there is no change in any failure point nor in any event acceptance criteria, there is no reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-4" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

ENCLOSURE 5A

MARK-UP OF NUREG-1431 SPECIFICATIONS

MARK-UP OF NUREG-1431 SPECIFICATIONS

Applicable Industry Travelers (1 Page)

Mark-up:

<u>SPECIFICATION</u>	<u>PAGE</u>
5.1	5.0-1
5.2	5.0-2
5.3	5.0-5
5.4	5.0-6
5.5	5.0-7
5.6	5.0-23
5.7	5.0-28
Methodology	(2 Pages)

Industry Travelers Applicable to CTS Section 6.0/ITS 5.0

TRAVELER #	STATUS	DIFFERENCE #	COMMENTS
TSTF-9, Rev. 1	Incorporated	B-PS	NRC Approved
TSTF-37, Rev. 1	Incorporated	5.6-2	DCPP only
TSTF-52	Incorporated	5.5-4	
TSTF-65	Not Incorporated	NA	Not NRC approved as of traveler cut-off date
TSTF-106, Rev. 1	Not Incorporated	NA	Retain CTS
TSTF-118	Incorporated	5.5-8	
TSTF-119	Not Incorporated	NA	Retain CTS
TSTF-120	Not Incorporated	NA	Retain CTS
TSTF-121	Incorporated	5.2-2	
TSTF-152	Incorporated	5.6-4	
TSTF-167	Incorporated	5.7-2	
WOG-67, Rev. 1	Incorporated	5.6-5	
WOG-72	Incorporated	5.5-13	
WOG-85	Incorporated	5.5-14	
Proposed Traveler	Incorporated	5.5-1	WOG min-group Action Item 147

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The ~~[Plant Superintendent]~~ Vice President, Diablo Canyon Operations and ~~Plant Manager~~, hereafter called Plant Manager, shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

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The ~~[Plant Superintendent]~~ ~~Plant Manager~~ or his designee, hereafter called ~~Plant Manager~~, shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

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5.1.2 The ~~[Shift Supervisor (SS)]~~ ~~Shift Foreman (SFM)~~ shall be responsible for the control room command function. During any absence of the ~~[SS]~~ ~~SFM~~ from the control room while the unit is in MODE 1, 2, 3, or 4, an individual ~~(other than the Shift Technical Advisor)~~ with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the ~~[SS]~~ ~~SFM~~ from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

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5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

- a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the ~~EF SAR Update~~; B
- b. The ~~[Plant Superintendent]~~ ~~Plant Manager~~ shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant; B-PS
- c. The ~~[a specified corporate executive position]~~ ~~Senior Vice President and General Manager, Nuclear Power Generation~~ shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and B-PS
- d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

- a. A non-licensed operator shall be assigned to each reactor containing fuel and ~~an additional~~ ~~with a total of three~~ non-licensed operators ~~required for both units~~.

(Continued)

5.2 Organization

5.2.2 Unit Staff (continued)

~~shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.~~

5.2-1

~~Two unit sites with both units shutdown or defueled require a total of three non licensed operators for the two units.~~

B-PS

~~b. At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.~~

eb. Shift crew composition may be one less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.g for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

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ec. A ~~health physics technician~~ shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

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ed. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), health physicists technicians, auxiliary nuclear operators, and key maintenance personnel).

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Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work an ~~8 or 12~~ hour day, nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant

B-PS

(Continued)

5.2 Organization

modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;

5.2 Organization

5.2.2 Unit Staff (continued)

2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;
3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;
4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized in advance by the ~~Plant Superintendent~~ ~~Plant Manager~~ or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the ~~Plant Superintendent~~ ~~Plant Manager~~ or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

OR

~~The amount of overtime worked by unit staff members performing safety related functions shall be limited and controlled in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).~~

- ~~f.~~ The ~~Operations Manager or Assistant Operations Manager~~ ~~Operations Director~~ shall hold an SRO license.
- g. The Shift Technical Advisor (STA) shall provide advisory technical support to the ~~SEM~~ in

B-PS

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5.2-6

5.2 Organization

the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. ~~in addition, the STA~~ The STA position shall be manned in MODES 1, 2, 3, and 4 unless an individual with a SRO license meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

Reviewer's Note: Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of unique organizational structures.

5.3.1 ~~Each member of the unit staff shall meet or exceed the minimum qualifications of [Regulatory Guide 1.8, Revision 2, 1987, or more recent revisions, or ANSI Standard acceptable to the NRC staff]. The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to NRC staff].~~

5.3-1

~~Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI/ANS 3-1-1978 for comparable positions, except for the Radiation Protection Director who shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 2, April 1987 for Radiation Protection Manager. The licensed ROs and SROs shall also meet or exceed the minimum qualifications of 10 CFR Part 55 and the supplemental requirements specified in Section A of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.~~

~~A retraining and replacement training program for the plant staff shall be maintained under the direction of a designated member of the facility staff and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.~~

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

- 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:
- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
 - b. The emergency operating procedures required to implement the applicable requirements of NUREG-0737 and ~~to~~ NUREG-0737, Supplement 1, as stated in ~~{Generic Letter 82-33 and response to the subject NUREGS}~~; ED
 - c. Quality assurance for effluent and environmental monitoring;
 - d. Fire Protection Program implementation; and
 - e. All programs specified in Specification 5.5.
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5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and
- b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification ~~[5.6.2]~~ and Specification ~~[5.6.3]~~.

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Licensee initiated changes to the ODCM:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
 - 1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and
 - 2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
- b. Shall become effective after the approval of the ~~[Plant Superintendent]~~ ~~Plant Manager~~; and
- c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the

B-PS

(Continued)

5.5 Programs and Manuals

5.5.1 Offsite Dose Calculation Manual (ODCM) (continued)

page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include ~~portions of Recirculation Spray, Safety Injection, Chemical and Volume Control, gas stripper, and Hydrogen Recombiner~~ Residual Heat Removal, RCS Sample, and Liquid and Gaseous Radwaste Treatment Systems. The program shall include the following:

B-PS

- a. Preventive maintenance and periodic visual inspection requirements; and
- b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases, iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

5.5-2

- a. Training of personnel;
- b. Procedures for sampling and analysis; and
- c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably

(Continued)

5.5 Programs and Manuals

achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to

(Continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;
- b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ~~10 times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001-20.2402~~^{10 CFR 20, Appendix B, Table 2, Column 2;} 5.5-1
- c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;
- d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;
- e. ~~Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days~~^{Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with methodology and parameters in the ODCM. Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.} 5.5-13
- f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;
- g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the 5.5-1

(Continued)

5.5 Programs and Manuals

site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1; conforming to the following:

1. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and [redacted]
 2. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ; [redacted]
- h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

(Continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

- i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and
- j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section [5.2 and 5.3], cyclic and transient occurrences to ensure that components are maintained within the design limits.

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~~5.5.6 Not Used~~

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~~5.5.6 Pre Stressed Concrete Containment Tendon Surveillance Program~~

~~This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.~~

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

(Continued)

5.5 Programs and Manuals (continued)

~~In lieu of Position C.4.b(1) and C.4.b(2), conduct a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or conduct a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels once every ten years coinciding with the Inservice Inspection schedule as required by ASME Section XI.~~

5.5-14

Reviewer's Note

~~1 Licensees shall confirm that the flywheels are made of SA-533-B material. Further, licensee having Group 15 flywheels (as determined in WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination") need to demonstrate that material properties of their A516 material is equivalent to SA-533-B material, and its reference temperature, RT_{min} , is less than 30°F.~~

~~2 For flywheels not made of SA-533-B or A516 material, licensees need to either demonstrate that the flywheel material properties are bounded by those of SA-533-B material, or provide the minimum specified ultimate tensile stress, the fracture toughness, and the reference temperature, RT_{min} , for that material. For the latter, the licensee should employ these material properties, and use the methodology in the topical report, as extended in the two responses to the staff's RAI, to provide an assessment to justify a change in inspection schedule for their plants.~~

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

5.5-3

- a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure
Vessel Code and
applicable Addenda

(Continued)

5.5 Programs and Manuals (continued)

<u>terminology for inservice testing activities</u>	<u>Required Frequencies for performing inservice testing activities</u>
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

- b. The provisions of SR 3.0.2 are applicable to the above required frequencies for performing inservice testing activities;
- c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and
- d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Tube Surveillance Program

~~Reviewer's Note: The Licensee's current licensing basis SG tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~

~~SG tube integrity shall be demonstrated by performance of the following augmented inservice inspection program:~~

5.5-8

~~The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test frequencies:~~

- a. ~~SG Sample Selection and Inspection - SG tube integrity shall be determined during shutdown by selecting~~

(Continued)

5.5 Programs and Manuals (continued)

and inspecting at least the minimum number of SGs specified in Table 5.5.9-1.

b) **SG Tube Sample Selection and Inspection** - The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of SG tubes shall be performed at the frequencies specified in Specification 5.5.9.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.d. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all SGs; the tubes selected for these inspections shall be selected on a random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each SG shall include:

a) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%);

b) Tubes in those areas where experience has indicated potential problems; and

c) A tube inspection (pursuant to Specification 5.5.9.d.1.h) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

3. The tubes selected as the second and third samples (if required by Table 5.5.9-2) during each inservice inspection may be subjected to a partial tube inspection provided:

a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found; and

(Continued)

5.5 Programs and Manuals (continued)

b) The inspections include those portions of the tubes where imperfections were previously found

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

c. Inspection Frequencies - The above required inservice inspections of SG tubes shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections not including the preservice inspection result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.
2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.5.9-2 at 40

(Continued)

5.5 Programs and Manuals (continued)

month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.c.1. The interval may then be extended to a maximum of once per 40 months; and

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

- a) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13; or
- b) A seismic occurrence greater than the Double Design Earthquake; or
- c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features; or
- d) A main steam line or feedwater line break.

d. Acceptance Criteria

1. As used in this Specification:

- a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.
- b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.
- c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.
- d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

(Continued)

5.5 Programs and Manuals (continued)

- e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.
 - f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.
 - g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of a Double Design Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9 c 3, above.
 - h) Tube Inspection means an inspection of the SG tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg; and
 - i) Preservice Inspection means an inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial Power Operation using the equipment and techniques expected to be used during subsequent inservice inspections.
2. The SG tube integrity shall be determined after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.9-2.

e. Reports

The contents and frequency of reports concerning the SG tube surveillance program shall be in accordance with Specification 5.6.10.

(Continued)

5.5 Programs and Manuals (continued)

TABLE 5.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS (SGs) TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
	Two	Three	Four	Two	Three	Four
No. of Steam Generators per Unit						
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One ¹			One ¹	One ²	One ³

TABLE NOTATIONS

1. The inservice inspection may be limited to one SG on a rotating schedule encompassing 3 N % of the tubes (where N is the number of SGs in the plant) if the results of the first or previous inspections indicate that all SGs are performing in a like manner. Note that under some circumstances, the operating conditions in one or more SGs may be found to be more severe than those in other SGs. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.
2. The other SG not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.
3. Each of the other two SGs not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

(Continued)

5.5 Programs and Manuals (continued)

TABLE 5.5.9-2

STEAM GENERATOR (SG) TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes Notification to NRC pursuant to §50.72(b)(2) of 10 CFR Part 50	N/A	N/A

$$S = 3 \frac{N}{n} \%$$

Where N is the number of SGs in the unit, and n is the number of SGs inspected during an inspection

(Continued)

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

5.5-12

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in ~~[Regulatory Guide 1.52]~~, below and in accordance with ~~[Regulatory Guide 1.52, Revision 2, ASME N510-1989 (1980), and AG-1 ASTM D3803-1989]~~.

~~(DCM S-23F, S-23B, and S-230)~~

- a. Demonstrate for each of the ESF systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ~~< [0.05 to 0.1%]~~ when tested in accordance with ~~[Regulatory Guide 1.52, Revision 2, and ASME N510-1989 (1980)]~~ at the system flowrate specified below ~~[10%]~~ at ~~least once per operating cycle.~~

(Continued)

5.5 Programs and Manuals (continued)

ESF Ventilation System	Flowrate
<div style="border: 1px solid black; padding: 5px; display: inline-block;"> Control Room Auxiliary Building Fuel Handling Building </div>	<div style="border: 1px solid black; padding: 5px; display: inline-block;"> 2100 cfm 73,500 cfm 35,750 cfm </div>

- b. Demonstrate for each of the ESF systems that an in-place test of the charcoal absorber absorber shows a penetration and system bypass < [0.05 1.0%] when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989 1980] at the system flowrate specified below [1 10%] at least once per operating cycle.

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ESF Ventilation System	Flowrate
<div style="border: 1px solid black; padding: 5px; display: inline-block;"> Control Room Auxiliary Building Fuel Handling Building </div>	<div style="border: 1px solid black; padding: 5px; display: inline-block;"> 2100 cfm 73,500 cfm 35,750 cfm </div>

- c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal absorber absorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of \leq [30°C] and greater than or equal to at the relative humidity specified below. Laboratory testing shall be completed at least once per 18 months and after every 720 hours of charcoal operation.

ED

ESF Ventilation System	Penetration	RH
<div style="border: 1px solid black; padding: 5px; display: inline-block;"> Control Room Auxiliary Building Fuel Handling Building </div>	<div style="border: 1px solid black; padding: 5px; display: inline-block;"> 1.0% 6.0% 4.3% </div>	<div style="border: 1px solid black; padding: 5px; display: inline-block;"> 70% 70% 95% </div>

(Continued)

5.5 Programs and Manuals (continued)

Reviewer's Note: Allowable penetration = ~~[100% methyl iodide efficiency for charcoal credited in staff safety evaluation]/ (safety factor).~~

Safety factor = ~~_____ [5] for systems with heaters.~~
~~_____ [7] for systems without heaters.~~

- d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, ~~the prefilters,~~ and the charcoal absorbers is less than the value specified below when tested in accordance with ~~[Regulatory Guide 1.52, Revision 2, and ASME N510-1989 1980]~~ at the system flowrate specified below ~~[± 10%]~~ at least once per operating cycle.

ED

ESF Ventilation System	Delta P	Flowrate
Control Room	3.5 in. WG	2100 cfm
Auxiliary Building	3.7 in. WG	73,500 cfm
Fuel Handling Building	4.1 in. WG	95,750 cfm

- e. Demonstrate that the charcoal pre-heaters for each of the ESF systems . dissipate the value specified below ~~[± 10%]~~ when tested in accordance with ~~[ASME N510-1989 1980]~~ at least once per operating cycle.

ESF Ventilation System	Wattage
Control Room	5 ± 1 KW
Auxiliary Building	50 ± 5 KW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

(Continued)

5.5 Programs and Manuals (continued)

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the ~~Waste Gas Holdup System~~, ~~the quantity of radioactivity contained in gas storage tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in temporary unprotected outdoor liquid storage tanks~~.

B-PS

5.5-9

The gaseous radioactivity quantities shall be determined following the methodology in ~~Branch Technical Position (BTP) ETSB 11.5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure"~~ Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure". The liquid radwaste quantities shall be determined in accordance with ~~Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures"~~ maintained such that ~~10 CFR Part 20~~ limits are met.

B-PS

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The program shall include:

- a. The limits for concentrations of hydrogen and oxygen in the ~~Waste Gas Holdup System~~ and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);
- b. A surveillance program to ensure that the quantity of radioactivity contained in ~~each gas storage tank and fed into the offgas treatment system~~ is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of ~~an uncontrolled release of the tanks' contents~~; and
- c. A surveillance program to ensure that the quantity of radioactivity contained in all temporary outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the ~~Liquid Radwaste Treatment System~~ is less

5.5-9

(Continued)

5.5 Programs and Manuals (continued)

than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2~~III~~, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
 - 1. an API gravity or an absolute specific gravity within limits,
 - 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
 - 3. a clear and bright appearance with proper color; ~~or water and sediment content within limits.~~ 5.5-5
- b. Other properties for ASTM 2D fuel oil are ~~within limits analyzed~~ within 31 days following sampling and addition to storage tanks; and 5.5-5
- c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 ~~or A-3.~~
- d. ~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.~~ 5.5-8

(Continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 - 1. a change in the TS incorporated in the license; or
 - 2. a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- d. Proposed changes that meet the criteria of Specification 5.5.14b, above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

- a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
- b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

5.5 Programs and Manuals (continued)

- c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
- d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

- a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
- b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program

5.5-4

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(c) and 10 CFR 50 Appendix J Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995."

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_s , is 47 psig.

The maximum allowable containment leakage rate, L_s , at P_s shall be 0.10 % of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq 1.0 L_s$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60 L_s$ for the Type B and Type C tests and $\leq 0.75 L_s$ for Type A tests.

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b. Air lock testing acceptance criteria are:

- 1) Overall air lock leakage rate is $\leq 0.05 L_s$ when tested at $\geq P_s$.
- 2) For each door, leakage rate is $\leq 0.01 L_s$ when pressurized to ≥ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 Backup Method for Determining Subcooling Margin

5.5-6

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

- 1) Training of personnel, and
- 2) Procedures for monitoring.

5.5 Programs and Manuals (continued)

5.5.18 Containment Polar and Turbine Building Cranes

5.5-6

A program which will ensure that: 1) the position of the containment polar cranes precludes jet impingement from a postulated pipe rupture, and 2) the operation of the turbine building cranes is consistent with the restrictions associated with the current Hosgri seismic analysis of the turbine building. This program shall include the following:

- 1) Training of personnel, and
 - 2) Procedures for the containment polar and turbine building cranes operation.
-
-

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

B

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man-rem exposure for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20% percent of the individual total dose need not be accounted for. In the aggregate, at least 80% percent of the total whole-body deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year. [The initial report shall be submitted by April 30 of the year following the initial criticality.]

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B-PS

B-PS

5.6.2 Annual Radiological Environmental Operating Report

B

-----NOTE-----

(Continued)

5.6 Reporting Requirements

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6-3

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements [in the a format of similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979]. [The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.] In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

B-PS

B-PS

5.6.3 Radioactive Effluent Release Report

B-PS

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

(Continued)

5.6 Reporting Requirements (continued)

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6-4

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

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5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

~~The individual specifications that address core operating limits must be referenced here.~~

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1. Shutdown Bank Insertion Limits for Specification 3.1.5
2. Control Bank Insertion Limits for Specification 3.1.6
3. Axial Flux Difference for Specification 3.2.3
4. Heat Flux Hot Channel Factor $K(z)$ and $W(z) = F_0(z) (F_0^{RR})$ Specification 3.2.1)
5. RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor F_{RH}^{N} (F_{RH}^{RR} and PF_{RH} for Specification 3.2.2)
6. SHUTDOWN MARGIN values in Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8

(Continued)

5.6 Reporting Requirements (continued)

7. Moderator Temperature Coefficient limits in Specification 3.1.3, and

8. Refueling Boron Concentration limits in Specification 3.9.1.

- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

B-PS

~~Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.~~

1. WCAP-10216-P-A, Revision 1A, Relaxation of Constant Axial Offset Control F, Surveillance Technical Specification, February 1994 (Westinghouse Proprietary)

2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (Westinghouse Proprietary)

3. WCAP-8385, Power Distribution Control and Load Following Procedures, September 1974 (Westinghouse Proprietary)

4. WCAP-10054-P-A, Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code, August 1985 (Westinghouse Proprietary) and

5. WCAP-10266-P-A, Revision 2 with Addenda, The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, December 14, 1987 (Westinghouse Proprietary)

- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(Continued)

5.6 Reporting Requirements (continued)

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing; Low Temperature Overpressure Protection (LTOP) arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following: ~~{The individual specifications that address RCS pressure and temperature limits must be referenced here.}~~

5.6-5

B-PS

1. ~~Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and~~
2. ~~Specification 3.4.12, "Low Temperature Overpressure Protection System"~~

- b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: ~~{Identify the NRC staff approval document by date.}~~

B-PS

~~1. The analytical methods used to determine the RCS pressure and temperature limits were developed in accordance with:~~

~~10 CFR 50, Appendix G and H
Regulatory Guide 1.99, Revision 2
NUREG-0800, Standard Review Plan Section 5.3.2
Branch Technical Position MTEB 5-2
ASME B&PV Code Section III, Appendix G
ASME B&PV Code, Section XI, Appendix A
WCAP-14040-NP-A, Section 2.2~~

~~2. LTOP limits (Power Operated Relief Valves (PORV) pressure relief setpoint and LTOP enable temperature) were developed in accordance with:~~

~~NUREG-0800, Standard Review Plan Section 5.2.2
Branch Technical Position RSB 5-2
10 CFR 50, Appendix G and H
Regulatory Guide 1.99, Revision 2
Branch Technical Position MTEB 5-2
WCAP-14040-NP-A, Section 2.2~~

(Continued)

5.6 Reporting Requirements (continued)

- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

~~Reviewers' Notes: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:~~

- ~~1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).~~
- ~~2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.~~
- ~~3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC approved methodologies may be included in the PTLR.~~
- ~~4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.~~
- ~~5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG 0800 Standard Review Plan 5.3.2, Pressure Temperature Limits.~~
- ~~6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.~~
- ~~7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{ref}) to the predicted increase in RT_{ref} ; where the predicted increase in RT_{ref} is based on the mean shift in RT_{ref} plus the two standard deviation value (2σ) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase $RT_{\text{ref}} + 2\sigma$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.~~

5.6-5

(Continued)

5.6 Reporting Requirements (continued)

~~g. LTOP arming temperature limit development methodology~~

5.6.7 EDG Failure Report ~~Not Used~~

5.6-2

~~If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.5, or existing Regulatory Guide 1.108 reporting requirement.~~

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

B

~~5.6.9~~ ~~Not Used~~

B-PS

~~5.6.9 Tendon Surveillance Report~~

~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.~~

(Continued)

5.6 Reporting Requirements (continued)

~~5.6.10 Steam Generator Tube Inspector Report~~

~~Reviewer's Note: Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used.~~

~~Reviewer's Note: These reports may be required covering inspection, test, and maintenance activities. These reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~

5.6.10 Steam Generator (SG) Tube Inspection Report

ED

B-PS

a. Within 15 days following the completion of each inservice inspection of SG tubes, the number of tubes plugged in each SG shall be reported to the Commission.

b. The complete results of the SG tube inservice inspection shall be submitted to the Commission in a report within 12 months following completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected;

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. Results of SG tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.0 ADMINISTRATIVE CONTROLS

~~{5.7 High Radiation Area}~~

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr at 30 cm, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., ~~{Health Physics Technicians}~~) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

5.7-1

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

- a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.
- b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.
- c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the ~~{radiation protection supervision}~~ in the RWP.

3-PS

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr at 30 cm shall be provided with locked or continuously guarded doors to prevent unauthorized inadvertent entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during

5.7-1

5.7-2

(Continued)

{5.7 High Radiation Area}

5.7.2 (continued)

periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

5.7.3

For individual high radiation areas with radiation levels of > 1000 mrem/hr at 30 cm, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

5.7-1

Methodology For Mark-up of NUREG-1431 Specifications

Enclosure 5A contains an electronic (or hand written) mark-up of NUREG-1431 Revision 1. The purpose of the mark-up is to identify those changes necessary to create a plant specific improved TS (by incorporating plant specific values in bracketed areas) and to identify any other changes with a cross-reference to a justification or explanation for the change. Descriptions/justifications for changes are contained in Enclosure 6A.

There are four types of changes:

1. Deletions - Material which is removed from NUREG-1431, Rev. 1.
2. Additions - This includes material which is added to NUREG-1431, Rev. 1.
3. Modifications - This includes material which exist in NUREG-1431, Rev. 1 but is being revised for the improved TS.
4. Bracket Inserts - These changes involve the insertion of plant specific information which is presently located in the current TS into a bracketed portion of NUREG-1431, Rev. 1.

The methodology of identifying the changes is:

- Deletions - The portion of the specification which is being deleted in non-bracketed areas of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand). The deletions are identified by a change number or a change code in the adjacent right margin.
- Additions - The information being added to the non-bracketed portions of NUREG-1431, Rev. 1 is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The addition is identified by a change number or a change code in the adjacent right margin.
- Modifications - The information being revised in the non-bracketed portions of NUREG-1431, Rev. 1 is annotated using the strike-out feature of WordPerfect (or crossed out by hand) and the revised information is inserted into the specification in the appropriate location and is annotated using the red-line feature of WordPerfect (or hand written/insert pages). The modification is identified by a change number or a change code in the adjacent right margin. A change code of "PS" indicates an obvious plant specific change and is usually reserved for plant specific names of systems and components.
- Editorial Changes- Changes/corrections which are obviously editorial are annotated using the red-line/strike-out feature of WordPerfect and identified by a change code of "Ed" in the adjacent margin. All such changes will be submitted for incorporation into the generic traveler for editorial changes.
- Bracket Inserts - The plant specific information is entered into the bracketed area. If "generic" information had been provided in the bracketed area and that information is not correct for this plant, the "generic" information is "struck-out" and the correct information inserted using the "red-line" feature. The brackets provided in NUREG-1431, Rev. 1 are deleted. "Red-line," "strike-out" and margin codes are as follows:
1. If the bracketed wording or parameter values remain unchanged, the bracketed information is "red-lined" and 'B' (for bracketed information) is used as the margin code.

**Methodology For Mark-up of NUREG-1431 Specifications
(Continued)**

2. If the bracketed wording or parameter values are changed to the plant specific wording/values in the current specifications, the old bracketed information is "struck-out," the new information is "red-lined" and 'B-PS' (for plant specific bracketed information) is used as a margin code.
3. If the entire Condition, Action, or Surveillance is bracketed and is applicable, the letter/number designator for the item is red-lined. The text included within the brackets is not red-lined unless plant specific changes are made. The 'B' or 'B-PS' margin code is used depending on whether plant specific changes were made.

If the entirely bracketed Condition/Action/Surveillance is not applicable, the entire contents are "struck-out," red-lined words "Not Used" are inserted, and a 'B-PS' margin code is used.

Changes which have margin identifiers of letters instead of numbers (i.e., B, B-PS, Ed or PS) do not have descriptions/justifications in Enclosure 6A.

Note: All brackets are removed as part of the mark-up process. Reviewer notes may be "struck-out" or deleted as preferred.

In summary, in the non-bracketed portions of NUREG-1431, Rev. 1, "red-line" is used to annotate new material, "strike-out" is used to annotate deleted material, and change numbers or change codes are used in the right margin to identify these changes. All changes (i.e., "red-line" or "strike-out" items) have a change number or a change code.

Note: NUREG-1431, Rev. 1 is used for all mark-ups. Industry Travelers which are incorporated are indicated using the "red-lines," "strike-outs" and margin codes discussed above.

ENCLOSURE 5B

Not Applicable

ENCLOSURE 6A

DIFFERENCES FROM NUREG-1431

Descriptions of NUREG-1431 Differences

(3 pages)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 5.0

This Enclosure contains a brief discussion/justification for each marked-up technical change to NUREG-1431, to make them plant-specific or to incorporate generic changes resulting from the Industry/NRC generic change process. The change numbers are referenced directly from the NUREG-1431 mark-ups (Enclosure 5A). For Enclosures 3A, 3B, 4, 6A, and 6B text in brackets "[]" indicates the information is plant specific and is not common to all the JLS plants. Empty brackets indicate that other JLS plants may have plant specific information in that location.

<u>CHANGE NUMBER</u>	<u>JUSTIFICATION</u>
5.1-1	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
5.1-2	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
5.2-1	This change revises Section 5.2.2a to reflect the CTS. This change clarifies the application of the unit staff provisions to both units.
5.2-2	This change deletes Section 5.2.2.b since the requirement for the presence of a RO or a SRO in the control room is adequately controlled by 10 CFR 50.54(m)(2)(iii) and 50.54(k). The ITS 5.2.2.b requirement that is being deleted will be met through compliance with these regulations and is not required in the TS. This is consistent with traveler TSTF-121.
5.2-3	Not Used.
5.2-4	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
5.2-5	Not Used.
5.2-6	This change revises section 5.2.2f to describe the current [TS] for the Shift Technical Advisor (STA).
5.2-7	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
5.3-1	This change revises Section 5.3.1 to be consistent with CTS regarding plant staff qualifications and training.
5.5-1	These changes revise Section 5.5.4, "Radioactive Effluent Controls Program," to reflect new 10 CFR Part 20 requirements and NRC letter dated 7/28/95 (Christopher I. Grimes to Owners Groups). A proposed traveler is being prepared to reflect changes required for NUREG-1431 to be consistent with 10 CFR Part 20.
5.5-2	This change revises Section 5.5.3, "Post Accident Sampling," to ensure the capability to obtain and analyze radioactive "iodines" in lieu of "gases." This change is consistent with the CTS and plant practices.
5.5-3	This change revises Section 5.5.8, "Inservice Testing Program," to delete "including applicable supports." This change is consistent with the CTS.
5.5-4	The Containment Leakage Rate Testing Program is added to the ITS consistent with the CTS. The Containment Leakage Rate Testing Program is consistent with traveler TSTF-52.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 5.0

CHANGE NUMBER

JUSTIFICATION

- 5.5-5 This change revises Section 5.5.13, "Diesel Fuel Oil Testing Program," to be consistent with the CTS. The details of the method applied to this test are discussed in the associated SR 3.8.3.3 Bases. [To maintain consistency with the Bases for 3.8.3.3, specific changes to the program description are for sampled properties of new fuel oil from "within limits" to "analyzed" within 31 days following sampling and addition of the fuel oil to storage tanks. This wording more clearly defines that within 31 days following initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties are met.]
- 5.5-6 Additional programs are added to the ITS (other than Containment Leakage Rate Testing Program discussed in CN 5.5-4). [The programs are: Backup Method for Determining Subcooling Margin (5.5.17) and Containment Polar and Turbine Building Cranes (5.5.18)].
- 5.5-7 Not applicable to DCCP. See Conversion Comparison Table (Enclosure 6B).
- 5.5-8 A sentence is added to Section 5.5.9 ("The provisions of SR 3.0.2 are applicable to the Steam Generator Tube Surveillance Program test frequencies") and Section 5.5.13 ("The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies") to provide consistency with current application of these requirements. This is consistent with the use of CTS and alleviates potential confusion in the program descriptions. This change is consistent with traveler TSTF-118.
- 5.5-9 This change revises Section 5.5.12c to clarify that temporary outdoor liquid radwaste tanks are covered under the surveillance program. This is consistent with current plant practices.
- 5.5-10 Not applicable to DCCP. See Conversion Comparison Table (Enclosure 6B).
- 5.5-11 Not applicable to DCCP. See Conversion Comparison Table (Enclosure 6B).
- 5.5-12 The referenced frequencies for the tests listed in the Ventilation Filter Testing Program (VFTP) were evaluated as part of the 24-month fuel cycle program (see LAR 96-09).
- 5.5-13 This change revises Radioactive Effluent Controls Program dose projections to meet original intent of TS prior to implementation of GL 89-01. GL 89-01 provided the wording for the STS (Section 5.5.4.e) which combined the requirements for cumulative and projected dose. This requires a plant to make projected doses for the quarter and year on a 31-day basis. It is only necessary and reasonable to make a projection for the next 31 days. A cumulative dose projection is still required for the current calendar quarter and year in accordance with the ODCM. This is consistent with WOG-72.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 5.0

CHANGE NUMBER

JUSTIFICATION

- 5.5-14 Section 5.5.7 is being revised consistent with WOG-85 []. The proposed changes to Section 5.5.7 provide an exception to the examination requirements in RG 1.14, Rev. 1, "Reactor Coolant Pump Flywheel Integrity." The proposed exception to the recommendations of Regulatory Position C.4.b would allow for an acceptable inspection method of either an ultrasonic volumetric or surface examination. The acceptable inspection method would be conducted at approximately 10 year intervals. This change is consistent with the NRC SER associated with WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination."
- 5.5-15 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 5.6-1 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).
- 5.6-2 This change deletes the Emergency Diesel Generator (EDG) Report to reflect the recommendations of GL 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," dated May 31, 1994.
- 5.6-3 This change revises the report date in Section 5.6.2, "Annual Radiological Environmental Operating Report," to be consistent with CTS.
- 5.6-4 This change revises Sections 5.6.1 and 5.6.3, "Occupational Radiation Exposure Report" and "Radioactive Effluent Release Report," respectively, per NRC letter dated July 28, 1995, "Changes to Technical Specifications Resulting from 10 CFR 20 and 50.36a Changes" (From Christopher I. Grimes to Owners Groups Chairs). This change is consistent with traveler TSTF-152.
- 5.6-5 [Low temperature overpressure protection (LTOP) arming temperature and] PORV lift settings are referenced in pressure and temperature limits report (PTLR) section per WOG-67, Rev. 1.
- 5.7-1 This change revises High Radiation Area to incorporate changes consistent with [10 CFR 20.1601]. Specifically, distances from the radiation source are noted.
- 5.7-2 This change revises "unauthorized" to "inadvertent" in the High Radiation Area section to reflect the NRC's position as stated in RG 8.38, Section 1.5 regarding physical barriers for high radiation areas. This is consistent with traveler TSTF-167.
- 5.7-3 Not applicable to DCP. See Conversion Comparison Table (Enclosure 6B).

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table

(5 Pages)

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.1-1	Revises Section 5.1.1 to maintain WCNOG CTS. The plant manager does not currently approve prior to implementation each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.	No	No	Yes	No
5.1-2	Revises Section 5.1.1 to maintain Callaway CTS that the plant manager approves prior to implementation each proposed test, experiment, or modification to systems or equipment that affect nuclear safety and are not addressed in the FSAR or TS.	No	No	No	Yes
5.2-1	Revises Section 5.2.2.a to reflect the CTS. This change clarifies the application of the unit staff provisions to both units.	Yes	Yes	No, Wolf Creek is a single unit site.	No, Callaway is a single unit site.
5.2-2	The requirement for the presence of a RO or an SRO in the control room may be deleted from the ITS since this requirement is adequately controlled by 10 CFR 50.54(m)(2)(iii). This change is consistent with traveler TSTF-121.	Yes	Yes	Yes	Yes
5.2-3	Not Used	NA	NA	NA	NA
5.2-4	Section 5.2.2.a describes the unit staff requirements for nonlicensed operator staffing for multi-unit sites. This change reflects plant specific requirements for a single unit site and is consistent with the CTS.	No, DCPD is a multi-unit plant.	No, CPSES is a multi-unit plant.	Yes	Yes
5.2-5	Not Used	NA	NA	NA	NA
5.2-6	Revises Section 5.2.2f to describe the current [TS] for STA.	Yes	Yes, LA 50/36 moved text to FSAR Section 13.1 which permits on-shift SRO to fill STA position.	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 5.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.2-7	Revises 5.2.2c to add note that a single radiation protection technician and a single chemistry technician may fulfill the requirements for both units.	No, not current procedure or operational requirement.	Yes	No, Wolf Creek is a single unit site.	No, Callaway is a single unit site.
5.3-1	Revises Section 5.3.1 to be consistent with CTS regarding plant staff qualifications and training.	Yes, LA 43/42.	Yes	Yes	Yes
5.5-1	Revises Section 5.5.4, "Radioactive Effluent Controls Program," to reflect new 10 CFR Part 20 requirements and NRC letter dated 7/28/95 consistent with proposed traveler.	Yes	Yes	Yes	Yes
5.5-2	Revises Section 5.5.3, "Post Accident Sampling," to ensure the capability to obtain and analyze radioactive "iodines" in lieu of "gases." This change is consistent with the CTS and plant practices.	Yes	Yes	Yes	Yes
5.5-3	Revises Section 5.5.8, "Inservice Testing Program," to delete "including applicable supports." This change is consistent with the CTS.	Yes	Yes	Yes	Yes
5.5-4	The Containment Leakage Rate Testing Program is added to the ITS consistent with the CTS. The Containment Leakage Rate Testing Program is consistent with traveler TSTF-52.	Yes, LA 110/109.	Yes, LAR 96-002.	Yes	Yes
5.5-5	Revises Section 5.5.13, "Diesel Fuel Oil Testing Program," to be consistent with CTS. The details of the methods applied to this test are discussed in the associated SR 3.8.3.3 Bases.	Yes	Yes	Yes	Yes
5.5-6	Additional programs are added to the ITS (other than Containment Leakage Rate Testing Program discussed in CN 5.5-4).	Yes	Yes	No, no additional programs added.	No, no additional programs added.

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 5.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-7	Revises Section 5.5.12c to delete the reference to 10 CFR 20, Appendix B, Table 2, Column 2 for Callaway and add reference to the Standard Review Plan (SRP), Section 15.7.3, Rev. 2. 10 CFR 20, Appendix B, Table 2, Column 2 is inconsistent with the guidance provided in the SRP, Section 15.7.3.	No	No	No	Yes
5.5-8	A sentence is added to Section 5.5.9 ("The provisions of SR 3.0.2 are applicable to the Steam Generator Tube Surveillance Program test frequencies") and Section 5.5.13 ("The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies") to provide consistency with current application of these requirements. This change is consistent with traveler TSTF-118.	Yes	Yes	Yes	Yes
5.5-9	Revises Section 5.5.12c to clarify that DCPD temporary outdoor liquid radwaste tanks are covered under the surveillance program. This is consistent with current plant practices.	Yes	No	No	No
5.5-10	This change lists the tanks that the surveillance program in Section 5.5.12c is applicable to as is in the CTS. This change is a plant specific requirement consistent with the CTS.	No, not part of CTS.	No, not part of CTS. Maintaining ITS wording.	Yes	Yes
5.5-11	The documents referenced for the testing frequency for the VFTP do not provide frequencies for combined pressure drop tests or the heater power rating test. The CTS frequency is added for these two tests.	No, see CN 5.5-12.	Yes	Yes	Yes
5.5-12	The referenced frequencies for the tests listed in the VFTP were evaluated as part of the DCPD 24 month fuel cycle program (see LAR 96-09).	Yes	No	No	No
5.5-13	Revises Radioactive Effluent Controls Program dose projections to meet original intent of TS prior to implementation of GL 89-01. (WOG-72)	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 5.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.5-14	Section 5.5.7 is being revised consistent with WOG-85 []. The proposed changes to Section 5.5.7 provide an exception to the examination requirements in RG 1.14, Rev. 1, "Reactor Coolant Pump Flywheel Integrity."	Yes	Yes	Yes	Yes
5.5-15	This change provides a time interval of within 31 days after removal in which a laboratory test of a sample obtained from the charcoal adsorber must be tested. This change is consistent with the Callaway CTS.	No	No	No	Yes
5.6-1	Revises Section 5.6.4, "Monthly Operating Report," to reflect a revised submittal date.	No, DCPD CTS consistent with NUREG-1431.	Yes, LAR 94-14.	No, Wolf Creek CTS consistent with NUREG-1431.	No, Callaway CTS consistent with NUREG-1431.
5.6-2	Deletes the EDG Report to reflect the recommendations of GL 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," dated May 31, 1994.	Yes	No, not in CTS.	No, not in CTS.	No, not in CTS.
5.6-3	Revises report date in Section 5.6.2, "Annual Radiological Environmental Operating Report," to be consistent with CTS.	Yes, consistent with CTS and LA 78/77.	Yes, see LA 42/28.	Yes	Yes
5.6-4	Revises Sections 5.6.1 and 5.6.3, "Occupational Radiation Exposure Report" and "Radioactive Effluent Release Report," respectively, per NRC letter dated July 28, 1995, "Changes to Technical Specifications Resulting from 10 CFR 20 and 50.36a Changes." (From Christopher I. Grimes to Owners Groups Chairs) This change is consistent with traveler TSTF-152.	Yes	Yes	Yes	Yes
5.6-5	[LTOP arming temperature and] PORV lift settings are referenced in PTLR section per WOG-67, Rev. 1.	Yes	Yes	Yes	Yes
5.7-1	Revises high radiation area to incorporate changes consistent with [10 CFR 20.1601].	Yes	Yes	Yes	Yes

CONVERSION COMPARISON TABLE FOR DIFFERENCES FROM NUREG-1431, SECTION 5.0

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
5.7-2	Changes "unauthorized" to "inadvertent" in the High Radiation Area section to reflect the NRC's position as stated in RG 8.38, Section 1.5, regarding physical barriers for High Radiation Areas. This change is consistent with traveler TSTF-167.	Yes	Yes	Yes	Yes
5.7-3	This change deletes the phrase "or that cannot be continuously guarded" from the ITS for Callaway to make them consistent with the CTS.	No	No	No	Yes

JLS Conversion to Improved Technical Specifications

Diablo Canyon Power Plant

Docket # 50-275

Accession # 9706230042

Date 6/2/97 of Ltr

Regulatory Docket File

Improved TS



1.0 USE AND APPLICATION

1.1 Definitions

-----NOTE-----

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of an excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass those components such as sensors, alarms, displays, and trip functions required to perform the specified safety function(s). Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION shall include an in-place cross calibration that compares the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation.

(continued)

1.1 Definitions (continued)

This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL
TEST (CFT)

A CFT shall be:

- a. Analog channels - the injection of a simulated or actual signal into the channel as close to the sensor as practical to verify OPERABILITY including required alarm and trip functions, or
- b. Bistable channels - the injection of a simulated or actual signal into the sensor to verify OPERABILITY including required alarm and trip functions, or
- c. Digital channels - the injection of a simulated or actual signal into the channel as close to the sensor input to the process racks as practical to verify OPERABILITY including required alarm and trip functions.

The Channel Functional Test may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel, is tested.

CHANNEL OPERATIONAL
TEST (COT)

A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY including all components in the channel such as alarms, interlocks, displays, and trip functions required to perform the specified safety function(s). The COT may be performed by means of any series of sequential, overlapping or total channel steps so that the entire channel is tested. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

(continued)

1.1 Definitions (continued)

CORE OPERATING LIMITS
REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of NRC Regulatory Guide 1.109, Rev. 1, October, 1977.

\bar{E} - AVERAGE
DISINTEGRATION ENERGY

\bar{E} shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 10 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY
FEATURE (ESF) RESPONSE
TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be verified by means of any series of sequential, overlapping, or total steps so that the entire response time is verified.

L_a

The maximum allowable primary containment leakage rate, L_a , shall be 0.10% of primary containment air weight per day at the calculated peak containment pressure (P_a).

(continued)

1.1 Definitions (continued)

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or
3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System.

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal

(continued)

1.1 Definitions (continued)

water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

- a. Described in Chapter 14 of the FSAR;
- b. Authorized under the provisions of 10 CFR 50.59; or
- c. Otherwise approved by the Nuclear Regulatory Commission.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in individual specifications.

QUADRANT POWER TILT RATIO (QPTR)

QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3338 Mwt for Unit 1 and 3411 Mwt for Unit 2.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be verified by means of any series of sequential,

(continued)

1.1 Definitions (continued)

- overlapping, or total steps so that the entire response time is verified.
- SHUTDOWN MARGIN (SDM) SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:
- a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and
 - b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.
- SLAVE RELAY TEST A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.
- STAGGERED TEST BASIS A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.
- THERMAL POWER THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
- TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) A TADOT shall consist of operating the trip actuating device and verifying OPERABILITY including all components, such as alarms, interlocks, displays, and trip functions required to perform the specified safety function(s). The TADOT may be performed by means of any series of sequential, overlapping or total channel steps so that the entire channel is tested. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

(continued)

1.1 Definitions (continued)

(continued)

Table 1.1-1 (page 1 of 1)

MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER ^(a)	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown ^(b)	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown ^(b)	< 0.99	NA	≤ 200
6	Refueling ^(c)	NA	NA	NA

(a) Excluding decay heat.

(b) The required reactor vessel head closure bolts fully tensioned.

(c) The required reactor vessel head closure bolts less than fully tensioned.

(continued)

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES The following examples illustrate the use of logical connectors.

(continued)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . <u>AND</u> A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

1.2 Logical Connectors

EXAMPLES
(continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Trip . . . <u>OR</u> A.2.1 Verify . . . <u>AND</u> A.2.2.1 Reduce . . . <u>OR</u> A.2.2.2 Perform . . . <u>OR</u> A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

(continued)

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition

(continued)

1.3 Completion Times

DESCRIPTION.
(continued)

However, when a subsequent train, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

- a. Must exist concurrent with the first inoperability; and
- b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

- a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or
- b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery" Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

(continued)

1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Function X train inoperable.	A.1 Restore Function X train to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One Function Y train inoperable.	B.1 Restore Function Y train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
C. One Function X train inoperable. <u>AND</u> One Function Y train inoperable.	C.1 Restore Function X train to OPERABLE status. <u>OR</u> C.2 Restore Function Y train to OPERABLE status.	72 hours 72 hours

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock". In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (including the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-5.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4.	12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

(continued)

1.3 Completion Times

EXAMPLE 1.3-6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Perform SR 3.x.x.x.	Once per 8 hours
	<u>OR</u> A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

EXAMPLES

EXAMPLE 1.3-6

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(continued)

1.3 Completion Times

EXAMPLES
(continued)

EXAMPLE 1.3-7.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time

(continued)

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-7 (continued)

Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE
COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

(continued)

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

EXAMPLES The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP <u>AND</u> 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>-----NOTE----- Not required to be performed until 12 hours after \geq 25% RTP. -----</p> <p>Perform channel adjustment.</p>	<p>7 days</p>

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches \geq 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power \geq 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>-----NOTE----- Only required to be performed in MODE 1. -----</p>	
<p>Perform complete cycle of the valve.</p>	<p>7 days</p>

The interval continues, whether or not the unit operation is in MODE 1, 2, or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance have been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply (as well as having had a violation of SR 3.0.4).

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify each containment isolation manual valve is closed.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days.

In Example 1.4-5, the "specified Frequency" begins when the Surveillance is performed, but when the interval expires the Surveillance is not required to be performed until certain conditions are met. The Surveillance must be performed prior to entering MODE 4 from MODE 5, but only if the 92 day "specified Frequency" has passed. Although the period prior to the specified conditions is given as 92 days, the time interval may be extended to 1.25 times the stated period as allowed by SR 3.0.2 for operational flexibility.

The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the conditions in the Frequency are met and the interval specified by SR 3.0.2 is exceeded without the Surveillance having been performed and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

(continued)

2.0 SAFETY LIMITS (SLS)

2.1 SLS

2.1.1 Reactor Core SLS

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLS specified in Figure 2.1.1-1.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained \leq 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

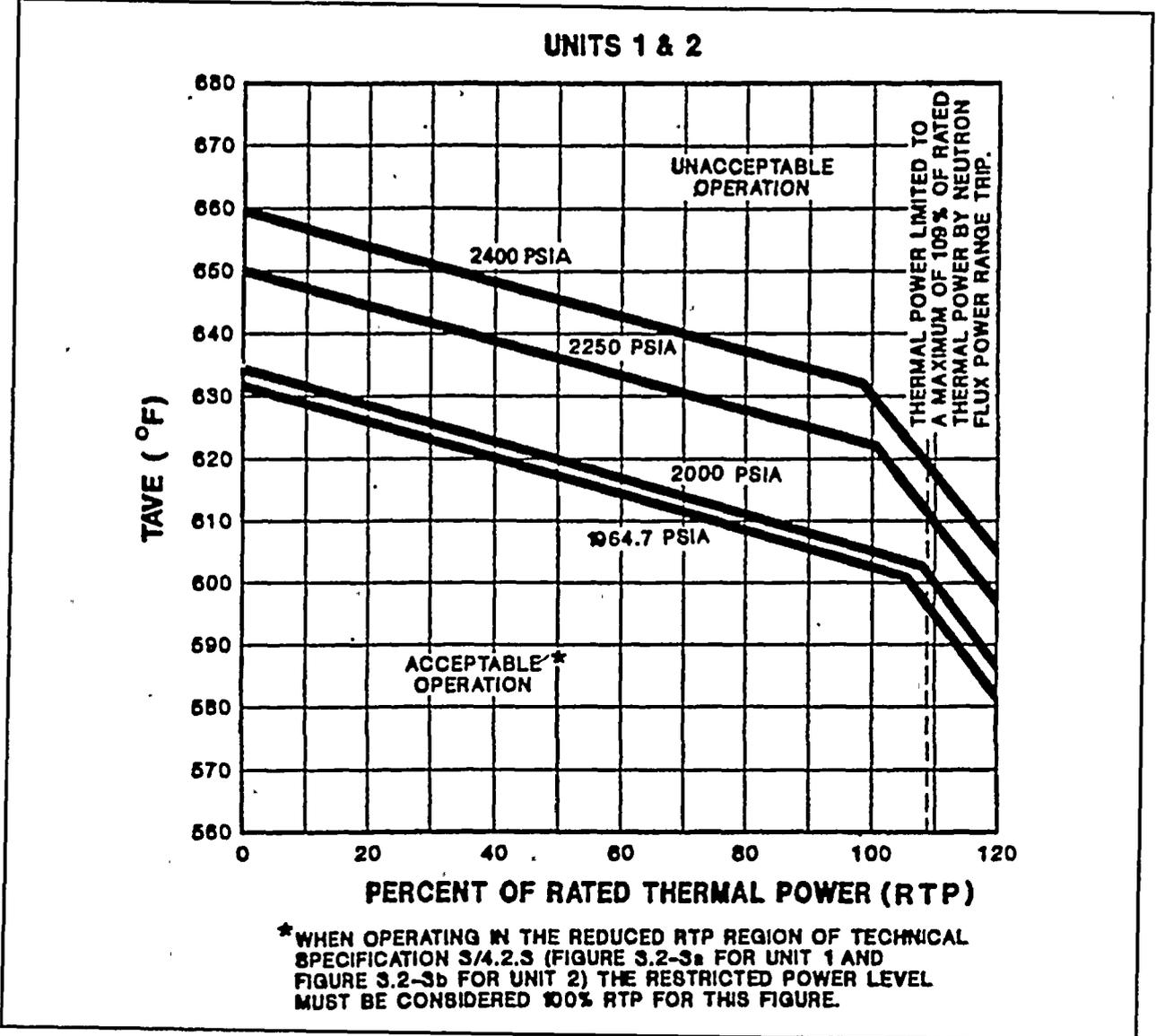


Figure 2.1.1-1

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

- a. MODE 3 within 7 hours;
- b. MODE 4 within 13 hours; and
- c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This LCO 3.0.4 Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications.

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.5 Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6 When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluations shall be performed in accordance with Specification 5.5.15, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

LCO 3.0.7 Test Exception LCO 3.1.8, allows specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1 SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2 The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply.

If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3 If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

SR 3.0.4 Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

(continued)

3.0 SR APPLICABILITY

SR 3.0.4
(continued)

SR 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3, and 4.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be within the limits provided in the COLR.

APPLICABILITY: MODE 2 with $k_{eff} < 1.0$,
MODES 3, 4, and 5.

-----NOTE-----
While this LCO is not met, entry into MODE 5 from MODE 6 is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM to be within limits.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within $\pm 1\%$ $\Delta k/k$ of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit.	A.1 Re-evaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation.	72 hours
	<u>AND</u> A.2 Establish appropriate operating restrictions and SRs.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.1.2.1 -----NOTE----- The predicted reactivity values shall be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. ----- Verify measured core reactivity is within $\pm 1\% \Delta k/k$ of predicted values.</p>	<p>Once prior to entering MODE 1 after each refueling <u>AND</u> -----NOTE----- Only required after 60 EFPD ----- 31 EFPD thereafter</p>

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR.
The maximum upper limit shall be that specified in Figure 3.1.3-1.

APPLICABILITY: MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ for the upper MTC limit,
MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within upper limit.	A.1 Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2 with $k_{eff} < 1.0$.	6 hours
C. MTC not within lower limit.	C.1 Be in MODE 4.	12 hours

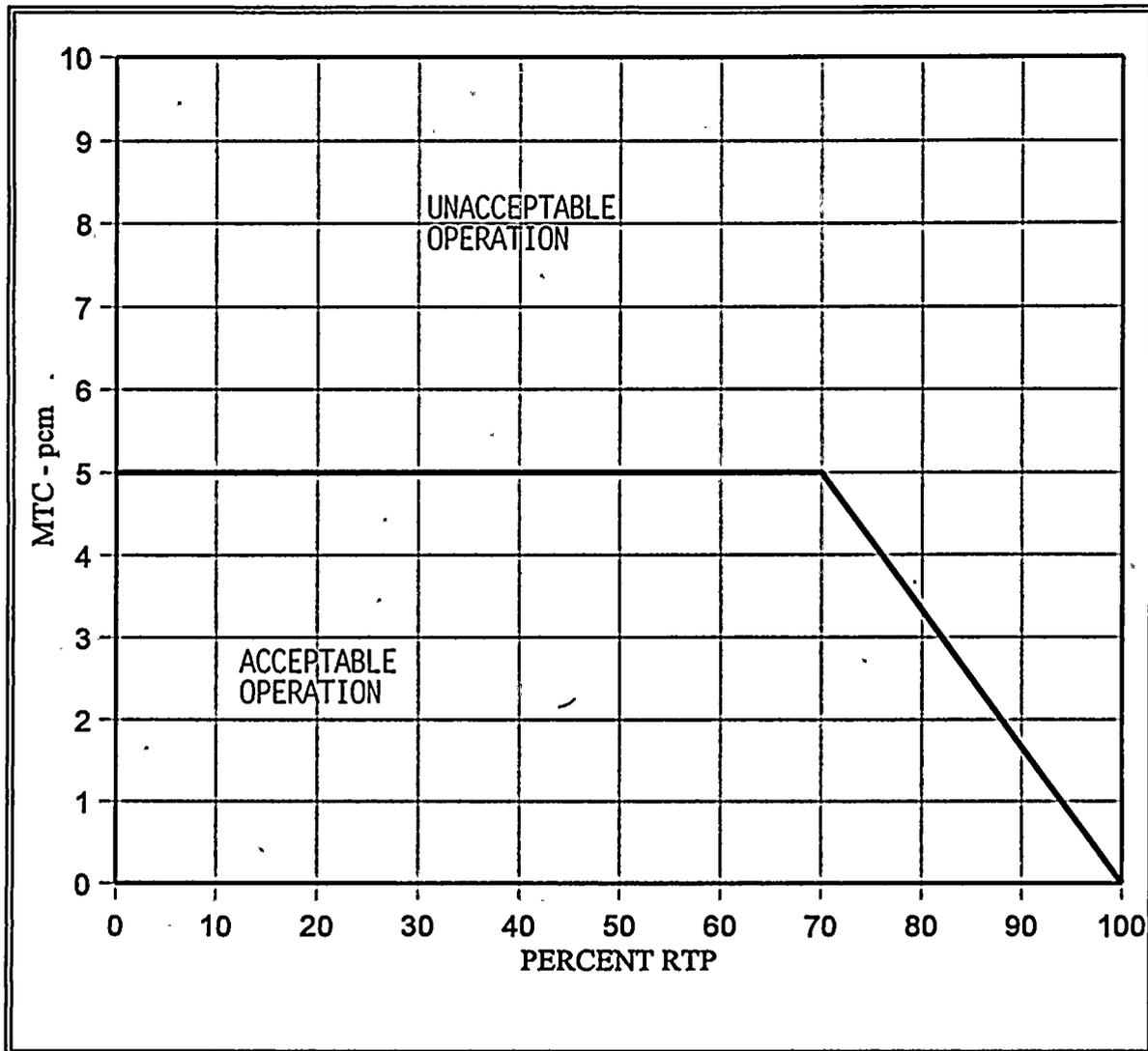


FIGURE 3.1.3-1 (page 1 of 1)
MODERATOR TEMPERATURE COEFFICIENT vs. POWER LEVEL

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE.

AND

Individual indicated rod positions shall be within 12 steps of their group step counter demand position.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1 Verify SDM to be within the limits provided in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Be in MODE 3.	6 hours
B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits.	1 hour
	<u>OR</u>	
	B.2.1.1 Verify SDM to be within the limits provided in the COLR.	1 hour
	<u>OR</u>	
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	B.2.2 Reduce THERMAL POWER to \leq 75% RTP.	2 hours
	<u>AND</u>	
	B.2.3 Verify SDM to be within the limits provided in the COLR.	Once per 12 hours
	<u>AND</u>	
	B.2.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	72 hours
	<u>AND</u>	
	B.2.5 Perform SR 3.2.2.1.	72 hours
	<u>AND</u>	
	B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. More than one rod not within alignment limit.	D.1.1 Verify SDM to be within the limits provided in the COLR.	1 hour
	<u>OR</u>	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	<u>AND</u>	
	D.2 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.7 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500^{\circ}\text{F}$; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.5 Each shutdown bank shall be within insertion limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with any control bank not fully inserted.

-----NOTE-----
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1 Verify SDM to be within the limits provided in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore shutdown banks to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 Verify each shutdown bank is within the limits specified in the COLR.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Bank Insertion Limits

LCO 3.1.6 Control banks shall be within the insertion, sequence, and overlap limits specified in the COLR.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

-----NOTE-----
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Control bank insertion limits not met.	A.1.1 Verify SDM to be within the limits provided in the COLR.	1 hour
	<u>OR</u>	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	<u>AND</u>	
	A.2 Restore control bank(s) to within limits.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Control bank sequence or overlap limits not met.	B.1.1 Verify SDM to be within the limits provided in the COLR. OR B.1.2 Initiate boration to restore SDM to within limit. AND B.2 Restore control bank sequence and overlap to within limits.	1 hour 1 hour 2 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify estimated critical control bank position is within the limits specified in the COLR.	Within 4 hours prior to achieving criticality
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	12 hours
SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors.	Once per 8 hours
	<u>OR</u> A.2 Reduce THERMAL POWER to \leq 50% RTP.	8 hours
B. More than one DRPI per group inoperable.	B.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors.	Once per 8 hours
	<u>AND</u> B.2 Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	Once prior to criticality after each removal of the reactor vessel head.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions - MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

- LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
- LCO 3.1.4, "Rod Group Alignment Limits";
- LCO 3.1.5, "Shutdown Bank Insertion Limits";
- LCO 3.1.6, "Control Bank Insertion Limits"; and
- LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

- a. RCS lowest operating loop average temperature is $\geq 531^{\circ}\text{F}$; and
- b. SDM is \geq within the limits provided in the COLR; and
- c. THERMAL POWER is $\leq 5\%$ RTP.

APPLICABILITY: During PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes
	<u>AND</u> A.2 Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit.	B.1 Open reactor trip breakers.	Immediately
C. RCS lowest operating loop average temperature not within limit.	C.1 Restore RCS lowest operating loop average temperature to within limit.	15 minutes

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1.	Within 12 hours prior to initiation of PHYSICS TESTS
SR 3.1.8.2	Verify the RCS lowest operating loop average temperature is $\geq 531^{\circ}\text{F}$.	30 minutes
SR 3.1.8.3	Verify THERMAL POWER is $\leq 5\%$ RTP.	1 hour
SR 3.1.8.4	Verify SDM is within the limits provided in the COLR.	24 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor (F₀(Z))

LCO 3.2.1 F₀(Z), as approximated by F₀^c(Z) and F₀^w(Z), shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. F ₀ ^c (Z) not within limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% F ₀ ^c (Z) exceeds limit.	15 minutes
	<u>AND</u>	
	A.2 Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% F ₀ ^c (Z) exceeds limit.	72 hours
	<u>AND</u>	
	A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% F ₀ ^c (Z) exceeds limit.	72 hours
	<u>AND</u>	
	A.4 Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. $F_0^W(Z)$ not within limits.	B.1 Reduce AFD limits $\geq 1\%$ for each 1% $F_0^W(Z)$ exceeds limit.	4 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
 During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify F ₀ ^c (Z) is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within 24 hours after achieving equilibrium conditions after exceeding, by ≥ 20% RTP, the THERMAL POWER at which F ₀ ^c (Z) was last verified <u>AND</u> 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.2.1.2 -----NOTE----- If F^c_q(Z) measurements indicate</p> <p style="text-align: center;">maximum over z $\left[\frac{F_q^c(z)}{K(z)} \right]$</p> <p>has increased since the previous evaluation of F^c_q(Z):</p> <p>a. Increase F^w_q(Z) by the appropriate factor specified in the COLR and reverify F^w_q(Z) is within limits; or</p> <p>b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate</p> <p style="text-align: center;">maximum over z $\left[\frac{F_q^c(z)}{K(z)} \right]$</p> <p>has not increased.</p> <p>-----</p> <p>Verify F^w_q(Z) is within limit.</p>	<p>Once after each refueling prior to THERMAL POWER exceeding 75% RTP</p> <p><u>AND</u></p> <p>(continued)</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	<p>Once within 24 hours after achieving equilibrium conditions after exceeding, by \geq 20% RTP, the THERMAL POWER at which F₀^H(Z) was last verified</p> <p><u>AND</u></p> <p>31 EFPD thereafter</p>

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor (F^N_{ΔH}).

LCO 3.2.2 F^N_{ΔH} shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ----- F^N_{ΔH} not within limit:</p>	<p>A.1.1 Restore F^N_{ΔH} within limit.</p>	4 hours
	<p><u>OR</u></p>	
	<p>A.1.2.1 Reduce THERMAL POWER to < 50% RTP.</p>	4 hours
	<p><u>AND</u></p>	
	<p>A.1.2.2 Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP.</p>	72 hours
	<p><u>AND</u></p>	
	<p>A.2 Perform SR 3.2.2.1.</p>	24 hours
	<p><u>AND</u></p>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.3</p> <p>-----NOTE----- THERMAL POWER does not have to be reduced to comply with this Required Action. -----</p> <p>Perform SR 3.2.2.1.</p>	<p>Prior to THERMAL POWER exceeding 50% RTP</p> <p><u>AND</u></p> <p>Prior to THERMAL POWER exceeding 75% RTP</p> <p><u>AND</u></p> <p>24 hours after THERMAL POWER reaching ≥ 95% RTP</p>
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----

During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify F ^N _{ΔH} is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD)

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

-----NOTE-----
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER \geq 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify AFD within limits for each OPERABLE excore channel.	7 days

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02 .

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. QPTR not within limit.</p>	<p>A.1 Reduce THERMAL POWER $\geq 3\%$ from RTP for each 1% of QPTR > 1.00.</p>	<p>2 hours</p>
	<p><u>AND</u></p>	
	<p>A.2 Determine QPTR and reduce THERMAL POWER $\geq 3\%$ from RTP for each 1% of QPTR > 1.00.</p>	<p>Once per 12 hours</p>
	<p><u>AND</u></p>	
	<p>A.3 Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.</p>	<p>24 hours</p>
<p><u>AND</u></p>		
<p>A.4 Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.</p>	<p>Once per 7 days thereafter</p>	
<p><u>AND</u></p>		
<p>A.4 Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.</p>	<p>Prior to increasing THERMAL POWER above the limit of Required Actions A.1 and A.2</p>	
<p><u>AND</u></p>		
	<p>(continued)</p>	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. (continued)</p>	<p>A.5 -----NOTE----- Perform Required Action A.5 only after Required Action A.4 is completed. -----</p> <p>Normalize excore detectors to eliminate tilt.</p> <p>AND</p> <p>A.6 -----NOTE----- Required Action A.6 must be completed when Required Action A.5 is implemented -----</p> <p>Perform SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1.</p>	<p>Prior to increasing THERMAL POWER above the limit of Required Actions A.1 and A.2</p> <p>Within 24 hours after reaching RTP</p> <p><u>OR</u></p> <p>Within 48 hours after increasing THERMAL POWER above the limit of Required Actions A.1 and A.2.</p>
<p>B. Required Action and associated Completion Time not met.</p>	<p>B.1 Reduce THERMAL POWER to \leq 50% RTP.</p>	<p>4 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.4.1</p> <p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER \leq 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. <p>-----</p> <p>Verify QPTR is within limit by calculation.</p>	<p>7 days</p>
<p>SR 3.2.4.2</p> <p>-----NOTE-----</p> <p>Not required to be performed until 12 hours after the input from one or more Power Range Neutron Flux channels is inoperable with THERMAL POWER $>$ 75% RTP.</p> <p>-----</p> <p>Verify QPTR is within limit using the movable incore detectors.</p>	<p>12 hours</p>

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LC0 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u> B.2 Be in MODE 3.	54 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>-----NOTE----- While this LCO is not met for function 19, 20 or 21, entry into MODE 5^(b) from MODE 5 is not permitted. This NOTE is an exception to the requirements of LCO 3.0.4. -----</p>		
<p>C. One channel or train inoperable.</p>	<p>C.1 Restore channel or train to OPERABLE status.</p> <p><u>OR</u></p> <p>C.2.1 Fully insert all rods.</p> <p><u>AND</u></p> <p>C.2.2 Place the Rod Control System in a condition incapable of rod withdrawal.</p>	<p>48 hours</p> <p>49 hours</p> <p>49 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. One Power Range Neutron Flux-High channel inoperable.</p>	<p>-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. -----</p>	
	<p>D.1.1 Place channel in trip. <u>AND</u></p>	<p>6 hours</p>
	<p>D.1.2 Reduce THERMAL POWER to \leq 75% RTP.</p>	<p>12 hours</p>
	<p><u>OR</u></p>	
	<p>D.2.1 Place channel in trip. <u>AND</u></p>	<p>6 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	<p>-----NOTE----- Not required to be performed until 12 hours after input from one Power Range Neutron Flux channel to QPTR is inoperable and THERMAL POWER is > than 75% RTP. -----</p> <p>D.2.2 Perform SR 3.2.4.2. <u>OR</u> D.3 Be in MODE 3.</p>	<p>Once per 12 hours</p> <p>12 hours</p>
E. One channel inoperable.	<p>-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels. -----</p> <p>E.1 Place channel in trip. <u>OR</u> E.2 Be in MODE 3.</p>	<p>6 hours</p> <p>12 hours</p>
F. One Intermediate Range Neutron Flux channel inoperable.	<p>F.1 Reduce THERMAL POWER to < P-6. <u>OR</u> F.2 Increase THERMAL POWER to > P-10.</p>	<p>24 hours</p> <p>24 hours</p>

(continued)

ACTIONS (continued)

CONDITION		REQUIRED ACTION	COMPLETION TIME
G.	Two Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions.	Immediately
		<u>AND</u> G.2 Reduce THERMAL POWER to < P-6.	2 hours
H.	Not used		
I.	One Source Range Neutron Flux channel inoperable.	I.1 Suspend operations involving positive reactivity additions.	Immediately
J.	Two Source Range Neutron Flux channels inoperable.	J.1 Open RTBs.	Immediately
K.	One Source Range Neutron Flux channel inoperable.	K.1 Restore channel to OPERABLE status.	48 hours
		<u>OR</u> K.2.1 Fully insert all rods.	49 hours
		<u>AND</u> K.2.2 Place the Control Rod System in a condition incapable of rod withdrawal.	49 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>L. Required Source Range Neutron Flux channel(s) inoperable.</p>	<p>L.1 Suspend operations involving positive reactivity additions.</p> <p><u>AND</u></p> <p>L.2 Perform SR 3.1.1.1.</p>	<p>Immediately</p> <p>1 hour</p> <p><u>AND</u></p> <p>Once per 12 hours thereafter</p>
<p>M. One channel inoperable.</p>	<p>-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. -----</p> <p>M.1 Place channel in trip.</p> <p><u>OR</u></p> <p>M.2 Reduce THERMAL POWER to < P-7.</p>	<p>6 hours</p> <p>12 hours</p>
<p>N. Not used</p>		
<p>O. Not used</p>		

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>P. One or more Turbine Trip channel(s) inoperable.</p>	<p>-----NOTE----- The inoperable low auto-stop oil pressure channel may be bypassed for up to 4 hours for surveillance testing of other channels. -----</p> <p>P.1 Place channel(s) in trip.</p> <p><u>OR</u></p> <p>P.2 Reduce THERMAL POWER to < P-9.</p>	<p>6 hours</p> <p>10 hours</p>
<p>Q. One train inoperable.</p>	<p>-----NOTE----- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. -----</p> <p>Q.1 Restore train to OPERABLE status.</p> <p><u>OR</u></p> <p>Q.2 Be in MODE 3.</p>	<p>6 hours</p> <p>12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>R. One RTB train inoperable.</p>	<p>-----NOTES-----</p> <p>1. One train may be bypassed for up to 2 hours for surveillance testing or maintenance, provided the other train is OPERABLE.</p> <p>2. One RTB may be bypassed only for the time required for performing maintenance on undervoltage or shunt trip mechanisms per CONDITION U, provided the other train is OPERABLE.</p> <p>3. One RTB may be bypassed for up to 4 hours for logic testing per CONDITION Q, provided the other train is OPERABLE.</p> <p>-----</p> <p>R.1 Restore train to OPERABLE status.</p> <p><u>OR</u></p> <p>R.2 Be in MODE 3.</p>	<p>1 hour</p> <p>7 hours</p>
<p>S. One or more required channels or trains inoperable.</p>	<p>S.1 Verify interlock is in required state for existing unit conditions.</p> <p><u>OR</u></p> <p>S.2 Be in MODE 3.</p>	<p>1 hour</p> <p>7 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>T. One or more required channels or trains inoperable.</p>	<p>T.1 Verify interlock is in required state for existing unit conditions.</p>	<p>1 hour</p>
	<p><u>OR</u> T.2 Be in MODE 2.</p>	<p>7 hours</p>
<p>U. One trip mechanism inoperable for one RTB.</p>	<p>U.1 Restore inoperable trip mechanism to OPERABLE status.</p>	<p>48 hours</p>
	<p><u>OR</u> U.2 Be in MODE 3.</p>	<p>54 hours</p>
<p>V. Not used</p>		

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>W. One channel inoperable</p>	<p>-----NOTE----- The inoperable channel may be bypassed for up to 72 hours for surveillance or maintenance. -----</p> <p>W.1 Place channel in trip</p>	<p>6 hours</p>
<p>X. One or more SG-low low Trip Time Delay circuit delay timers. inoperable.</p>	<p>X.1 Adjust the Trip Time Delay threshold power level for 0 seconds time delay to 0% RTP.</p> <p><u>OR</u></p> <p>X.2 Place the affected SG-low low level in trip.</p>	<p>6 hours</p> <p>6 hours</p>

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply for each RTS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 24 hours after THERMAL POWER is \geq 15% RTP, but prior to exceeding 30% RTP. <p>-----</p> <p>Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.</p>	24 hours
SR 3.3.1.3	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Adjust NIS channel if absolute difference is \geq 3%. 2. Not required to be performed until 24 hours after THERMAL POWER is \geq 50% RTP. <p>-----</p> <p>Compare results of the incore detector measurements to NIS AFD.</p>	31 effective full power days (EFPD)

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.4 -----NOTE----- This Surveillance must be performed on the reactor trip bypass breaker, for the local manual shunt trip only, prior to placing the bypass breaker in service. ----- Perform TADOT.</p>	<p>31 days on a STAGGERED TEST BASIS</p>
<p>SR 3.3.1.5 Perform ACTUATION LOGIC TEST.</p>	<p>31 days on a STAGGERED TEST BASIS</p>
<p>SR 3.3.1.6 -----NOTE----- Not required to be performed until 24 hours after achieving equilibrium conditions with THERMAL POWER \geq 75% RTP. ----- Calibrate excore channels to agree with incore detector measurements.</p>	<p>92 EFPD</p>
<p>SR 3.3.1.7 -----NOTE----- 1. Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. 2. For source range instrumentation, this Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. ----- Perform COT.</p>	<p>92 days</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.1.8 -----NOTE----- This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. -----</p> <p>Perform COT.</p>	<p>-----NOTE----- Only required when not performed within previous 92 days -----</p> <p>Prior to reactor startup</p> <p><u>AND</u></p> <p>12 hours after reducing power below P-10 for power and intermediate instrumentation</p> <p><u>AND</u></p> <p>Four hours after reducing power below P-6 for source range instrumentation</p> <p><u>AND</u></p> <p>Every 92 days thereafter</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	<p>-----NOTE----- Verification of setpoint is not required. -----</p> <p>Perform TADOT.</p>	92 days
SR 3.3.1.10	<p>-----NOTE----- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. -----</p> <p>Perform CHANNEL CALIBRATION.</p>	18 months
SR 3.3.1.11	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Neutron detectors are excluded from CHANNEL CALIBRATION. 2. This Surveillance shall include verification that the time constants are adjusted to the prescribed values. 3. Power and Intermediate Range detector plateau voltage verification is not required to be performed prior to entry in to MODE 2 or 1.. <p>-----</p> <p>Perform CHANNEL CALIBRATION.</p>	18 months
SR 3.3.1.12	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.13	Perform COT.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.14	<p>-----NOTE----- Verification of setpoint is not required. -----</p> <p>Perform TADOT.</p>	18 months
SR 3.3.1.15	<p>-----NOTE----- Verification of setpoint is not required. -----</p> <p>Perform TADOT.</p>	<p>-----NOTE----- Only required when not performed within previous 31 days -----</p> <p>Prior to reactor startup</p>
SR 3.3.1.16	<p>-----NOTE----- Neutron detectors are excluded from response time testing. -----</p> <p>Verify RTS RESPONSE TIME is within limits.</p>	18 months on a STAGGERED TEST BASIS
SR 3.3.1.17	Perform ACTUATION LOGIC TEST.	18 months

Table 3.3.1-1 (page 1 of 10)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Manual Reactor Trip	1.2	2	B	SR 3.3.1.14	NA	NA
	3 ^(b) , 4 ^(b) , 5 ^(b)	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						≤ 109% RTP
a. High	1.2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 111.1% RTP	≤ 25% RTP
b. Low	1 ^(c) , 2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 27.1% RTP	
3. Power Range Neutron Flux Rate						
a. High Positive Rate	1.2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.5% RTP with time constant ≥ 2 sec	≤ 5% RTP with time constant ≥ 2 sec
b. High Negative Rate	1.2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 6.5% RTP with time constant ≥ 2 sec	≤ 5% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1 ^(c) , 2 ^(d)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.9% RTP	≤ 25% RTP

(continued)

- (a) Not used
- (b) With Rod Control System capable of rod withdrawal or all rods not fully inserted.
- (c) Below the P-10 (Power Range Neutron Flux) interlocks.
- (d) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

Table 3.3.1-1 (page 2 of 10)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
5. Source Range Neutron Flux	2 ^(e)	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 1.4 E5 cps	≤ 1.0 E5 cps
	3 ^(b) , 4 ^(b) , 5 ^(b)	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16.	≤ 1.4 E5 cps	≤ 1.0 E5 cps
	3 ^(f) , 4 ^(f) , 5 ^(f)	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	N/A
6. Overtemperature ΔT	1.2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1 (Page 3.3-24)	Refer to Note 1 (Page 3.3-24)
7. Overpower ΔT	1.2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 2 (Page 3.3-25)	Refer to Note 2 (Page 3.3-25)

(continued)

- (a) Not used
- (b) With Rod Control System capable of rod withdrawal or all rods not fully inserted.
- (e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
- (f) With the RTBs open or all rods fully inserted and incapable of withdrawal. In this condition, source range Function does not provide reactor trip but does provide indication.

Table 3.3.1-1 (page 3 of 10)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
8. Pressurizer Pressure						
a. Low	1 ^(g)	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1944.4 psig	≥ 1950 psig
b. High	1.2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2390.6 psig	≤ 2385 psig
9. Pressurizer Water Level - High	1 ^(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 92.5%	≤ 92%
10. Reactor Coolant Flow - Low	1 ^(g)	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10SR 3.3.1.16	≥ 89.7% ⁽¹⁾	≥ 90% ⁽¹⁾

(continued)

(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

(1) Minimum measured flow is 89.800 gpm per loop for Unit 1 and 90.625 gpm per loop for Unit 2.

Table 3.3.1-1 (page 4 of 10)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
11. Reactor Coolant Pump (RCP) Breaker Position	1(g)	1 per RCP	M	SR 3.3.1.14	NA	NA
12. Undervoltage RCPs	1(g)	2 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 7730 V- each bus	≥ 8050 V- each bus
13. Underfrequency RCPs	1(g)	3 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 53.9 Hz-each bus	≥ 54.0 Hz-each bus
14. Steam Generator (SG) Water Level - Low Low	1.2	3 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 6.8%	≥ 7.2%
Coincident with:						
a) RCS Loop ΔT equivalent to power \leq 50% RTP with a time delay (TD).	1.2	4 (1/loop)	X	SR 3.3.1.7 SR 3.3.1.10	RCS loop ΔT variable input \leq 51.5% RTP	RCS loop ΔT variable input \leq 50%
OR					\leq (1.01) TD (Note 3)	\leq TD (Note 3)
b) RCS Loop ΔT equivalent to power $>$ 50% RTP with no time delay.	1	4(1/loop)	X	SR 3.3.1.7 SR 3.3.1.10	RCS loop ΔT variable input $>$ 51.5 RTP TD=0	RCS loop ΔT variable input $>$ 50% RTP TD=0
15. Not used						

(continued)

(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page6 of10)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
16. Turbine Trip						
a. Low Auto- Stop Oil Pressure	1(j)	3	P	SR 3.3.1.10 SR 3.3.1.15	≥ 45 psig	≥ 50 psig
b. Turbine Stop Valve Closure	1(j)	4	P	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open	≥ 1% open
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1.2	2 trains	Q	SR 3.3.1.14	NA	NA
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux. P-6	2(e)	2	S	SR 3.3.1.11 SR 3.3.1.13	≥ 6E-11 amp	≥ 1E-10 amp
b. Low Power Reactor Trips Block. P-7	1	1 per train	T	NA	NA	NA
c. Power Range Neutron Flux. P-8	1	3	T	SR 3.3.1.11 SR 3.3.1.13	≤ 37.1% RTP	≤ 35% RTP
d. Power Range Neutron Flux. P-9	1	3	T	SR 3.3.1.11 SR 3.3.1.13	≤ 52.1% RTP	≤ 50% RTP
e. Power Range Neutron Flux. P-10	1.2	3	S	SR 3.3.1.11 SR 3.3.1.13	≥ 7.9% RTP and ≤ 12.1% RTP	≥ 10% RTP
f. Turbine Impulse Chamber Pressure. P-13	1	2	T	SR 3.3.1.10 SR 3.3.1.13	≤ 12.1% RTP turbine impulse pressure equivalent	≤ 10% RTP turbine impulse pressure equivalent

(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(j) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 8 of 10)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
19. Reactor Trip Breakers ^(k)	1.2	2 trains	R	SR 3.3.1.4	NA	NA
	3 ^(b) , 4 ^(b) , 5 ^(b)	2 trains	C	SR 3.3.1.4	NA	NA
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms ^(k)	1.2	1 each per RTB	U	SR 3.3.1.4	NA	NA
	3 ^(b) , 4 ^(b) , 5 ^(b)	1 each per RTB	C	SR 3.3.1.4	NA	NA
21. Automatic Trip Logic	1.2	2 trains	Q	SR 3.3.1.5	NA	NA
	3 ^(b) , 4 ^(b) , 5 ^(b)	2 trains	C	SR 3.3.1.5	NA	NA
22. Seismic Trip	1.2	3 directions (x,y,z) in 3 locations	W	SR 3.3.1.12 SR 3.3.1.14 SR 3.3.1.17	≤ 0.40 _y	≤ 0.35 _y

(b) With Rod Control System capable of rod withdrawal or all rods not fully inserted.

(k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 9 of 10)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.0% of ΔT span.

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} \leq \Delta T_o \left\{ K_1 - K_2 \left[\frac{(1 + \tau_1 s)}{(1 + \tau_2 s)} T - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT , °F.
 ΔT_o is the indicated ΔT at RTP, °F.
 s is the Laplace transform operator, sec⁻¹.
 T is the measured RCS average temperature, °F.
 T' is the nominal T_{avg} at RTP, = 576.6 (Unit 1) & 577.6 (Unit 2)°F.

P is the measured pressurizer pressure, psig
 P' is the nominal RCS operating pressure, = 2235 psig

$K_1 \leq 1.20$ $K_2 = 0.0182/^\circ\text{F}$ $K_3 = 0.000831/\text{psig}$
 $\tau_1 \geq 30 \text{ sec}$ $\tau_2 \leq 4 \text{ sec}$
 $\tau_4 \geq 0 \text{ sec}$ $\tau_5 \leq 0 \text{ sec}$

$f_1(\Delta I) =$
 - 0.0275{ 19 + ($q_t - q_b$) } when $q_t - q_b \leq -19\%$ RTP
 0% of RTP when -19% RTP < $q_t - q_b \leq 7\%$ RTP
 0.0238{ ($q_t - q_b$) - 7 } when $q_t - q_b > 7\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 10 of 10)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 1.0% of ΔT span.

$$\Delta T \frac{(1 + \tau_4 s)}{(1 + \tau_5 s)} \leq \Delta T_O \left\{ K_4 - K_5 \frac{\tau_3 s}{1 + \tau_3 s} T - K_6 [T - T'] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT , °F.
 ΔT_O is the indicated ΔT at RTP, °F.
 s is the Laplace transform operator, sec⁻¹.
 T is the measured RCS average temperature, °F.
 T' is the nominal T_{avg} at RTP, ≤ 576.6 (Unit 1) & 577.6 (Unit 2)°F.

$K_4 \leq 1.072$	$K_5 \geq 0.0174/^\circ\text{F}$ for increasing T_{avg} $0/^\circ\text{F}$ for decreasing T_{avg}	$K_6 \geq 0.00145/^\circ\text{F}$ when $T > T'$ $0/^\circ\text{F}$ when $T \leq T'$
$\tau_1 \geq 30$ sec	$\tau_2 \leq 4$ sec	$\tau_3 \leq 10$ sec
$\tau_4 \leq 0$ sec	$\tau_5 \geq 0$ sec	

$f_2(\Delta I) = 0\%$ RTP for all ΔI .

Note 3: Steam Generator Water Level Low-Low Trip Time Delay

$$TD = B1(P)^3 + B2(P)^2 + B3(P) + B4$$

Where: P = RCS Loop ΔT Equivalent to Power (%RTP). P $\leq 50\%$ RTP

TD = Time delay for Steam Generator Water Level Low-Low Reactor Trip (in seconds).

B1 = -0.007128
 B2 = +0.8099
 B3 = -31.40
 B4 = +464.1

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or train(s).	Immediately
B. One channel or train inoperable.	B.1 Restore channel or train to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	B.2.2 Be in MODE 5.	84 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. One train inoperable.</p>	<p>C.1 -----NOTE----- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----- Restore train to OPERABLE status.</p> <p><u>OR</u></p> <p>C.2.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>C.2.2 Be in MODE 5.</p>	<p>6 hours</p> <p>12 hours</p> <p>42 hours</p>
<p>D. One channel inoperable.</p>	<p>D.1 -----NOTE----- The inoperable channel or one additional channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----- Place channel in trip.</p> <p><u>OR</u></p> <p>D.2.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2.2 Be in MODE 4.</p> <p><u>AND</u></p> <p>D.2.3 Be in MODE 5 for Function 1.c.</p>	<p>6 hours</p> <p>12 hours</p> <p>18 hours</p> <p>42 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. One Containment Pressure channel inoperable.</p>	<p>E.1 -----NOTE----- One additional channel may be bypassed for up to 4 hours for surveillance testing. -----</p> <p>Place channel in bypass.</p> <p><u>OR</u></p> <p>E.2.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>E.2.2 Be in MODE 4.</p> <p><u>AND</u></p> <p>E.2.3 Be in MODE 5 for Functions 2.c and 3.b.(3).</p>	<p>6 hours</p> <p>12 hours</p> <p>18 hours</p> <p>42 hours</p>
<p>F. One channel or train inoperable.</p>	<p>F.1 Restore channel or train to OPERABLE status.</p> <p><u>OR</u></p> <p>F.2.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>F.2.2 Be in MODE 4.</p>	<p>48 hours</p> <p>54 hours</p> <p>60 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G. One train inoperable.</p>	<p>G.1 -----NOTE----- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----- Restore train to OPERABLE status.</p> <p><u>OR</u></p> <p>G.2.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>G.2.2 Be in MODE 4.</p>	<p>6 hours</p> <p>12 hours</p> <p>18 hours</p>
<p>H. One train inoperable.</p>	<p>H.1 -----NOTE----- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----- Restore train to OPERABLE status.</p> <p><u>OR</u></p> <p>H.2 Be in MODE 3.</p>	<p>6 hours</p> <p>12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. One channel inoperable.	I.1 -----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ----- Place channel in trip. <u>OR</u> I.2.1 Be in MODE 2. <u>AND</u> 1.2.2 Be in MODE 3 for function 5.b.	 6 hours 12 hours 12 hours
J. Not used		
K. Not used.		

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>L. One or more required channels or train inoperable.</p>	<p>L.1 Verify interlock is in required state for existing unit condition.</p> <p><u>OR</u></p> <p>L.2.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>L.2.2 Be in MODE 4.</p>	<p>1 hour</p> <p>7 hours</p> <p>13 hours</p>
<p>M. One RCS Loop Delta-T channel inoperable.</p>	<p>M.1 Adjust the Trip Time Delay threshold power level for zero seconds time delay to 0% RTP.</p> <p><u>OR</u></p> <p>M.2 Place the affected SG water level low-low channel in trip.</p>	<p>6 hours</p> <p>6 hours</p>
<p>N. One channel inoperable.</p>	<p>N.1 Restore channel to OPERABLE status.</p> <p><u>OR</u></p> <p>N.2.1 Declare associated pump or valve inoperable.</p> <p><u>AND</u></p> <p>N.2.2 Comply with REQUIRED ACTION of 3.7.5 or 3.7.2 as applicable.</p>	<p>48 hours</p> <p>Immediately</p> <p>Immediately</p>

SURVEILLANCE REQUIREMENTS

-----NOTE-----
 Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS
 Function.

SURVEILLANCE	FREQUENCY
SR 3.3.2.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2 Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.3 Not used	
SR 3.3.2.4 Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.5 Perform COT.	92 days
SR 3.3.2.6 Perform SLAVE RELAY TEST.	18 months
SR 3.3.2.7 Not Used	

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.3.2.8 -----NOTE----- Verification of setpoint not required for manual initiation functions. ----- Perform TADOT.</p>	<p>18 months</p>
<p>SR 3.3.2.9 -----NOTE----- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. ----- Perform CHANNEL CALIBRATION.</p>	<p>18 months</p>
<p>SR 3.3.2.10 -----NOTE----- Not required to be performed for the turbine driven AFW pump until 24 hours after SG pressure is \geq 650 psig. ----- Verify ESFAS RESPONSE TIMES are within limit.</p>	<p>18 months on a STAGGERED TEST BASIS</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.2.11 -----NOTE----- Verification of setpoint not required. ----- Perform TADOT.	18 months

Table 3.3.2-1 (page 1 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1.2.3.4	2	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1.2.3.4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High	1.2.3.4	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 3.3 psig	≤ 3.0 psig
d. Pressurizer Pressure - Low	1.2.3 ^(b)	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 1844.4 psig	≥ 1850 psig
e. Steam Line Pressure						
(1)Low	1.2.3 ^(b)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 594.6 ^(c) psig	≥ 600 ^(c) psig
(2)Not used						
f. Not used						

(continued)

(a) Not used.

(b) Above the P-11 (Pressurizer Pressure) interlock. Trip function may be blocked in this MODE below the P-11 (pressurizer interlock) setpoint.

(c) Time constants used in the lead/lag controller are $t_1 = 50$ seconds and $t_2 = 5$ seconds.

(d) Not used.

(e) Not used.

(f) Not used.

Table 3.3.2-1 (page 2 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
g. Not used						
2. Containment Spray						
a. Manual Initiation	1.2.3.4	2 with, 2 coincident switches	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1.2.3.4.	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure						
High-High.	1.2.3.4	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 22.3 psig	≤ 22 psig
Not used						

(continued)

Table 3.3.2-1 (page 3 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
3. Containment Isolation						
a. Phase A Isolation						
(1) Manual Initiation	1.2.3.4	2	B	SR 3.3.2.8	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1.2.3.4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
b. Phase B Isolation						
(1) Manual Initiation	1.2.3.4	2 with 2 coincident switches	B	SR 3.3.2.8	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1.2.3.4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(3) Containment Pressure						
High-High	1.2.3.4	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 22.3 psig	≤ 22 psig
4. Steam Line Isolation						
a. Manual Initiation						
(1) Manual Initiation	1.2 ⁽¹⁾ , 3 ⁽¹⁾	1/valve	N	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays						
(1) Automatic Actuation Logic and Actuation Relays	1.2 ⁽¹⁾ , 3 ⁽¹⁾	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA

(continued)

(1) Except when all MSIVs are closed and de-activated.

Table 3.3.2-1 (page 4 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation (continued)						
c. Containment Pressure - High -High	1.2(i) 3(i)	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 22.3 psig	≤ 22.0 psig
d. Steam Line Pressure						
(1) Low	1.2(i) 3(b)(i)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 594.6 (c) psig	≥ 600(c) psig
(2) Negative Rate - High	3(g)(i)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	(h) 105.4 psi/sec	≤ 100 (h) psi/sec
e. Not used						
f. Not used						
g. Not used						
h. Not used						

(continued)

- (b) Above the P-11 (Pressurizer Pressure) interlock. Trip function may be blocked in this MODE below the P-11 (pressurizer interlock) setpoint.
- (c) Time constants used in the lead/lag controller are $t_1 = 50$ seconds and $t_2 = 5$ seconds
- (g) Trip function automatically blocked above P-11 (Pressurizer Pressure Interlock) setpoint and is automatically blocked below P-11 when Safety Injection on Steam Line Pressure-Low is not blocked.
- (h) Time constant utilized in the rate/lag controller are $t_1 = 50$ sec and $t_2 = 50$ sec.
- (i) Except when all MSIVs are closed and de-activated.

Table 3.3.2-1 (page 5 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1.2(j)	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. SG Water Level - High High (P-14)	1.2(j)	3 per SG	I	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 75.5%	≤ 75%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater						
a. Manual	1.2.3	1 sw/pp	N	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1.2.3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Not used						

(continued)

Table 3.3.2-1 (page 6 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Auxiliary Feedwater (continued)						
d. SG Water Level - Low Low	1.2.3 ^(k)	3 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 6.8%	≥ 7.2%
Coincident with:						
1) RCS Loop ΔT Equivalent to Power ≤ 50% RTP	1.2	4(1/loop)	D	SR 3.3.2.5 SR 3.3.2.9	RCS Loop ΔT Variable input ≤ 51.5% RTP	RCS Loop ΔT variable input ≤ 50% RTP
With a time delay (TD)	1.2	4 (1/loop)	M	SR 3.3.2.5 SR 3.3.2.9	≤ (1.01) TD (1)	≤ TD (1)
<u>Or</u>						
2) RCS Loop ΔT Equivalent to Power > 50% RTP	1.2	4(1/loop)	D	SR 3.3.2.5 SR 3.3.2.9	RCS Loop ΔT variable input > 51.5% RTP	RCS Loop ΔT variable input > 50% RTP
With no time delay	1.2	4 (1/loop)	M	SR 3.3.2.5 SR 3.3.2.9	TD=0	TD=0
(continued)						

(j) Except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

(k) For Mode 3, the Trip Time Delay associated with the Steam Generator Water Level-Low-Low channel must be less than or equal to 464.1 seconds.

(l) Steam Generator Water Level Low-Low Trip Time Delay

$$TD = B1(P)^3 + B2(P)^2 + B3(P) + B4$$

Where: P = RCS Loop ΔT Equivalent to Power (%RTP), P ≤ 50% RTP

TD = Time delay for Steam Generator Water Level Low-Low (in seconds)

$$B1 = -0.007128$$

$$B2 = +0.8099$$

$$B3 = -31.40$$

$$B4 = +464.1$$

Table 3.3.2-1 (page 7 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
6. Auxiliary Feedwater (continued)						
e. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
f. Not used						
g. Undervoltage Reactor Coolant Pump	1	2 per bus	I	SR 3.3.2.8 SR 3.3.2.9 SR 3.3.2.10	≥ 7730 volts	≥ 8050 volts
h. Not used						
i. Not used						
7. Residual Heat Removal Pump Trip on Refueling Water Storage Tank (RWST) Level-low	1.2.3.4	3	K	SR 3.3.2.1 SR 3.3.2.8 SR 3.3.2.9	≤ 32.9%	≤ 33%

(continued)

Table 3.3.2-1 (page 8 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
7. Not used						
8. ESFAS Interlocks						
a. Reactor Trip. P-4	1.2.3	1 per train. 2 trains	F	SR 3.3.2.11	NA	NA
b. Pressurizer Pressure. P-11	1.2.3	2	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1920.6 psig	≤ 1915 psig
c. Not used						

3.3 INSTRUMENTATION

3.3.3 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.3 The PAM instrumentation for each Function in Table 3.3.3-1 shall be OPERABLE.

APPLICABILITY MODES 1, 2 and 3.

ACTIONS

- NOTES-----
1. LCO 3.0.4 is not applicable.
 2. Separate Condition entry is allowed for each Function.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one required channel inoperable, but at least one valid channel OPERABLE.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.8.	Immediately
C. -----NOTE----- Not applicable to hydrogen monitor channels. ----- One or more Functions with no required channels OPERABLE.	C.1 Restore one channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION
<p>-----NOTE----- Only applicable in MODES 1 and 2 -----</p> <p>D. Two hydrogen monitor channels inoperable.</p>	<p>D.1 Restore one hydrogen monitor channel to OPERABLE status.</p>	<p>72 hours</p>
<p>E. Required Action and associated Completion Time of Condition C or D not met.</p>	<p>E.1 Enter the Condition referenced in Table 3.3.3-1 for the channel.</p>	<p>Immediately</p>
<p>F. As required by Required Action E.1 and referenced in Table 3.3.3-1.</p>	<p>F.1 Be in MODE 3. <u>AND</u> F.2 Be in MODE 4.</p>	<p>6 hours 12 hours</p>
<p>G. As required by Required Action E.1 and referenced in Table 3.3.3-1.</p>	<p>G.1 Initiate action in accordance with Specification 5.6.8.</p>	<p>Immediately</p>
<p>H. As required by Required Action E.1 and referenced in Table 3.3.3-1.</p>	<p>H.1 Be in MODE 3</p>	<p>6 hours</p>

SURVEILLANCE REQUIREMENTS

-----NOTE-----
 SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function
 in Table 3.3.3-1.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.3.2	<p>-----NOTE-----</p> <ol style="list-style-type: none"> 1. Neutron detectors are excluded from CHANNEL CALIBRATION. 2. CHANNEL CALIBRATION for Containment Area Radiation may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source. <p>-----</p> Perform CHANNEL CALIBRATION.	18 months

Table 3.3.3-1 (page 1 of 2)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITION REFERENCED FROM REQUIRED ACTION E.1
1. Neutron Flux (Wide Range NIS)	2	F
2. Steam Line Pressure	2 per steam generator	F
3. Reactor Coolant System (RCS) Hot Leg Temperature-- T_{hot} (Wide Range)	2 (1 per loop in two loops)	F
4. RCS Cold Leg Temperature-- T_{cold} (Wide Range)	2 (1 per loop in two loops)	F
5. RCS Pressure (Wide Range)	2	F
6. Reactor Vessel Water Level Indication System	2	G
7. a) Containment Recirculation Sump Water Level (Narrow Range)	2	F
b) Containment Reactor Cavity Sump Level--Wide Range	2	F
8. a) Containment Pressure (Wide Range)	2	F
b) Containment Pressure (Normal Range)	2	F
9. Containment Isolation Valve Position	2 per penetration flow path (a) (b)	F
10. Containment Area Radiation (High Range)	2	G
11. Hydrogen Monitors	2	H
12. Pressurizer Level	2	F
13. a) Steam Generator Water Level (Wide Range)	1 per steam generator	F
b) Steam Generator Water Level (Narrow Range)	2 per steam generator	F
14. Condensate Storage Tank Level	2	F
15. Incore Thermocouples - Quadrant 1	2 ^(c) per core quadrant	F
16. Incore Thermocouples- Quadrant 2	2 ^(c) per core quadrant	F

(continued)

Table 3.3.3-1 (page 2 of 2)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITION REFERENCED FROM REQUIRED ACTION E.1
17. Incore Thermocouples- Quadrant 3	2 ^(c) per core quadrant	F
18. Incore Thermocouples- Quadrant 4	2 ^(c) per core quadrant	F
19. Auxiliary Feedwater Flow	1 per steam generator	
20. Refueling Water Storage Tank Water Level	2	F

- (a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.
- (b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.
- (c) A channel consists of two Incore thermocouples.

3.3 INSTRUMENTATION

3.3.4 Remote Shutdown System

LCO 3.3.4 The Remote Shutdown System Instrumentation Functions and the SD panel controls in Table 3.3.4-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTIONS

- NOTES-----
1. LCO 3.0.4 is not applicable.
 2. Separate Condition entry is allowed for each Function.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable.	A.1 Restore required Function to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND B.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.4.1 -----NOTE----- Reactor Trip Breaker position is excluded from CHANNEL CHECK ----- Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.4.2 Verify each required control circuit and transfer switch is capable of performing the intended function.	18 months
SR 3.3.4.3 -----NOTE----- Reactor Trip Breaker position is excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION for each required instrumentation channel.	18 months
SR 3.3.4.4 Not used	

Table 3.3.4-1 (page 1 of 1)
Remote Shutdown System Instrumentation and Controls

	FUNCTION/INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF FUNCTIONS
1.	Reactivity Control	
	a. Not used	
	b. Reactor Trip Breaker Position	1 per trip breaker
	c. Not used	
2.	Reactor Coolant System (RCS) Pressure Control	
	a. Pressurizer Pressure	1
	b. Not used	
3.	Decay Heat Removal	
	a. RCS Hot Leg Temperature (loop 1 only)	1
	b. RCS Cold Leg Temperature (loop 1 only)	1
	c. AFW Controls	2 of any 3 pumps
	d. SG Pressure	1 per SG
	e. SG Level Wide Range	1 per SG
	or	
	AFW Flow	
	f. Condensate Storage Tank Level	1
	g. Component Cooling Water Control	Any 2 of 3 pumps
	h. Auxiliary Saltwater Control	2 of 2 pumps
	i. Emergency Diesel Generator Control	3 of 3 diesel generators
4.	RCS Inventory Control	
	a. Pressurizer Level	1
	b. Charging Pump Controls	2 of 2 pumps
	c. Charging Flow	1

3.3 INSTRUMENTATION

3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

LCO 3.3.5 One channel per bus of the loss of voltage DG start Function and two channels for initiation of load shed Function and two channels per bus of the degraded voltage Function with one timer per bus for DG start and initiation of load shed Function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.
When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more Functions with one or more channels per bus inoperable.</p>	<p>A.1 -----NOTE----- One channel may be bypassed for up to 2 hours for surveillance testing. ----- Declare the associated DG inoperable and enter the applicable Condition(s) and Required Action(s).</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Not used	
SR 3.3.5.2	Perform TADOT.	18 months
SR 3.3.5.3	<p>Perform CHANNEL CALIBRATION with Trip Setpoint and Allowable Value as follows:</p> <p>a. Loss of voltage Diesel Start Trip Setpoint and Allowable Value ≥ 0 V with a time delay of ≤ 0.8 second and ≥ 2583 V with a ≤ 10 second time delay.</p> <p>Loss of voltage initiation of load shed with one relay Trip Setpoint and Allowable Value ≥ 0 V with a time delay of ≤ 4 seconds and ≥ 2583 V with a time delay ≤ 25 seconds and with one relay ≥ 2870, instantaneous.</p> <p>b. Degraded voltage Diesel Start Trip Setpoint and Allowable Value ≥ 3785 V with a time delay of ≤ 10 seconds.</p> <p>Degraded voltage initiation of Load Shed Trip Setpoint and Allowable Value ≥ 3785 V with a time delay of ≤ 20 seconds.</p>	18 months

3.3 INSTRUMENTATION

3.3.6 Containment Purge and Exhaust Isolation Instrumentation

LCO 3.3.6 The Containment Purge and Exhaust Isolation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6-1

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One radiation monitoring channel inoperable.	A.1 Restore the affected channel to OPERABLE status.	4 hours
<p>B. -----NOTE----- Only applicable in MODE 1, 2, 3, or 4. -----</p> <p>One or more Functions with one or more manual or automatic actuation trains inoperable.</p> <p><u>OR</u></p> <p>Both radiation monitoring channels inoperable.</p> <p><u>OR</u></p> <p>Required Action and associated Completion Time of Condition A not met.</p>	B.1 Enter applicable Conditions and Required Actions of LCO 3.6.3, "Containment Isolation Valves," for containment purge and exhaust isolation valves made inoperable by isolation instrumentation.	Immediately

(continued)

SURVEILLANCE REQUIREMENTS

-----NOTE-----
 Refer to Table 3.3.6-1 to determine which SRs apply for each
 Containment Purge and Exhaust Isolation Function.

SURVEILLANCE	FREQUENCY
SR 3.3.6.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.6.2 Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.6.3 Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.6.4 Perform CFT.	92 days
SR 3.3.6.5 Perform SLAVE RELAY TEST.	18 months
SR 3.3.6.6 Not used	
SR 3.3.6.7 Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.8 Verify ESF Containment Purge and Exhaust Isolation response time is within limits.	18 months on a STAGGERED TEST BASIS

Containment Purge and Exhaust Isolation Instrumentation
3.3.6

Table 3.3.6-1 (page 1 of 1)
Containment Purge and Exhaust Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Not used				
2. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4, 6, and (a)	2 trains	SR 3.3.6.2 SR 3.3.6.3 SR 3.3.6.5	NA
3. Containment Radiation	1, 2, 3, 4, and (a)			
a. Gaseous 44 A/B	1, 2, 3, 4 (a)	2 1	SR 3.3.6.1 SR 3.3.6.4 SR 3.3.6.7 SR 3.3.6.8	Per ODCM
4. Containment Isolation - Phase A	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a., for all initiation functions and requirements.			

(a) During CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

3.3 INSTRUMENTATION

3.3.7 Control Room Ventilation System (CRVS) Actuation Instrumentation

LCO 3.3.7 The CRVS actuation instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	A.1 Place one CRVS train in pressurization mode.	7 days
B. One or more Functions with two channels or two trains inoperable.	B.1.1 Place one CRVS train in emergency pressurization mode.	Immediately
	<u>AND</u> B.1.2 Enter applicable Conditions and Required Actions for one CRVS train made inoperable by inoperable CRVS actuation instrumentation.	Immediately
C. Required Action and associated Completion Time for Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 5.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time for Condition A or B not met during movement of irradiated fuel assemblies	D.1 Suspend CORE ALTERATIONS.	Immediately
	AND D.2 Suspend movement of irradiated fuel assemblies.	Immediately
E. Required Action and associated Completion Time for Condition A or B not met in MODE 5 or 6.	E.1 Initiate action to restore one CRVS train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.7-1 to determine which SRs apply for each CRVS Actuation Function.

SURVEILLANCE	FREQUENCY
SR 3.3.7.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2 Perform CFT.	92 days
SR 3.3.7.3 Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.7.4 Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.7.5 Perform SLAVE RELAY TEST.	18 months
SR 3.3.7.6 -----NOTE----- Verification of setpoint is not required. ----- Perform TADOT.	18 months
SR 3.3.7.7 Perform CHANNEL CALIBRATION.	18 months

Table 3.3.7-1 (page 1 of 1)
CRVS Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1, 2, 3, 4, 5, 6, and (a)	2 trains	SR 3.3.7.6	NA
2. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4, 5, 6, and (a)	2 trains	SR 3.3.7.3 SR 3.3.7.4 SR 3.3.7.5	NA
3. Control Room Radiation	1, 2, 3, 4, 5, 6, and (a)			
a. Control Room Atmosphere Air Intakes	1, 2, 3, 4, 5, 6, and (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.7	≤ 2 mR/hr
4. Safety Injection			Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.	

(a) During movement of irradiated fuel assemblies.

3.3 INSTRUMENTATION

3.3.8 Fuel Building Ventilation System (FBVS) Actuation Instrumentation

LCO 3.3.8 The FBVS actuation instrumentation for each Function in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.8-1.

ACTIONS

- NOTE-----
1. Separate Condition entry is allowed for each Function.
 2. LCO 3.0.3 is not applicable
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more channels or trains inoperable.	A.1.1 Restore the inoperable monitors to OPERABLE status.	30 days
	<u>AND</u>	
	A.1.2.1 Install an appropriate portable continuous monitor with the same alarm setpoint.	Immediately
	<u>OR</u>	
	A.1.2.2 Station an individual qualified in radiation protection procedures with a dose rate monitoring device in the spent fuel pool area.	Immediately
	<u>OR</u>	
	A.1.2.3.1 Place one FBVS train in the Iodine Removal mode.	Immediately
	<u>AND</u>	
	A.1.2.3.2 Enter applicable Conditions and Required Actions of LCO 3.7.13, "Fuel Building Air Cleanup System (FBACS)," for one train made inoperable by inoperable actuation instrumentation.	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Not used		
C. Required Action and associated Completion Time for Condition A not met during movement of irradiated fuel assemblies in the fuel building.	C.1 Suspend movement of irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.8-1 to determine which SRs apply for each FBVS Actuation Function.

SURVEILLANCE	FREQUENCY
SR 3.3.8.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.8.2 Perform CFT.	92 days
SR 3.3.8.3 Not used	
SR 3.3.8.4 -----NOTE----- Verification of setpoint is not required. ----- Perform TADOT.	18 months
SR 3.3.8.5 Perform CHANNEL CALIBRATION.	18 months

Table 3.3.8-1 (page 1 of 1)
FBACS Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	(a)	2	SR 3.3.8.4	NA
2. Not used				
3. Fuel Building Radiation				
a. Spent Fuel Pool ^(b)	(a)	1	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.5	≤ 75 mR/hr
b. New Fuel Storage Vault ^(b)	(a)	1	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.5	≤ 15 mR/hr
c. Gaseous ^(c)	(a)	2	SR 3.3.8.1 SR 3.3.8.2 SR 3.3.8.5	per ODCM

(a) During movement of irradiated fuel assemblies in the fuel building.

(b) The requirements for FHBV mode change will not be applicable to the spent fuel storage pool or new fuel storage vault monitors following the installation of the gaseous monitors RM-45A and 45B.

(c) The requirements for FHBV mode change are applicable following the installation of RM-45A and 45B.

3.3 INSTRUMENTATION

3.3.9 Boron Dilution Protection System (BDPS) - Not used

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

- a. Pressurizer pressure \geq 2197.3 psig;
- b. RCS average temperature \leq 584.3°F; and
- c. RCS total flow rate within limits shown on Table 3.4.1-1 for Unit 1 and Table 3.4.1-2 for Unit 2.

APPLICABILITY: MODE 1.

-----NOTE-----
 Pressurizer pressure limit does not apply during:
 a. THERMAL POWER ramp > 5% RTP per minute; or
 b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.1.1 Verify pressurizer pressure is \geq 2197.3 psig.	12 hours
SR 3.4.1.2 Verify RCS average temperature is \leq 584.3°F.	12 hours
SR 3.4.1.3 Verify RCS total flow rate is within limits.	12 hours
SR 3.4.1.4 Verify measured RCS total flow rate is within limits.	18 months

Table 3.4.1-1
Reduction in Percent RATED THERMAL POWER for Reduced RCS Flow Rate
Unit 1

RCS Total Flow ^(a) (10 ⁴ GPM)	Acceptable Operating Region ^(b) (% RTP)
≥ 35.9	≤ 100%
≥ 35.6	≤ 98%
≥ 35.2	≤ 96%
≥ 34.8	≤ 94%
≥ 34.5	≤ 92%
≥ 34.1	≤ 90%

(a) For RCS Total Flow < 341,000 GPM, entry into LCO 3.4.1 Condition A is required.

(b) When operating in the restricted power limits, the restricted power level shall be considered 100% RTP for Figure 2.1.1-1.

Table 3.4.1-2

Reduction in Percent RATED THERMAL POWER for Reduced RCS Flow Rate
Unit 2

RCS Total Flow ^(a) (10 ⁴ GPM)	Acceptable Operating Region ^(b) (% RTP)
≥ 36.3	≤ 100%
≥ 35.9	≤ 98%
≥ 35.5	≤ 96%
≥ 35.2	≤ 94%
≥ 34.8	≤ 92%
≥ 34.4	≤ 90%

(a) For RCS Total Flow < 344,000 GPM, entry into LCO 3.4.1 Condition A is required.

(b) When operating in the restricted power limits, the restricted power level shall be considered 100% RTP for Figure 2.1.1-1.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each operating RCS loop average temperature (T_{avg}) shall be $\geq 541^\circ\text{F}$.

APPLICABILITY: MODE 1.
MODE 2 with $k_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more operating RCS loops not within limit.	A.1 Be in MODE 2, with $K_{eff} < 1.0$	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} in each operating loop $\geq 541^\circ\text{F}$.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Required Action A.2 shall be completed whenever this Condition is entered. ----- Requirements of LCO not met in MODE 1, 2, 3, or 4.</p>	<p>A.1 Restore parameters to within limits. <u>AND</u> A.2 Determine RCS is acceptable for continued operation.</p>	<p>30 minutes 72 hours</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5 with RCS pressure < 500 psig.</p>	<p>6 hours 36 hours</p>
<p>C. -----NOTE----- Required Action C.2 shall be completed whenever this Condition is entered. ----- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.</p>	<p>C.1 Initiate action to restore parameter(s) to within limits. <u>AND</u> C.2 Determine RCS is acceptable for continued operation.</p>	<p>Immediately Prior to entering MODE 4</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.3.1 -----NOTE----- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ----- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.</p>	<p>30 minutes</p>



3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODES 1 and 2

LCO 3.4.4 Four RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify each RCS loop is in operation.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE, and either:

- a. Two RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or
- b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

-----NOTE-----
All reactor coolant pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:

- a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
-

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours
C. One required RCS loop not in operation, with Rod Control System capable of rod withdrawal.	C.1 Restore required RCS loop to operation.	1 hour
	<u>OR</u> C.2 Place the Rod Control System in a condition incapable of rod withdrawal.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Four RCS loops inoperable. <u>OR</u> No RCS loop in operation.	D.1 Place the Rod Control System in a condition incapable of rod withdrawal.	Immediately
	<u>AND</u> D.2 Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	<u>AND</u> D.3 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.5.1 Verify required RCS loops are in operation.	12 hours
SR 3.4.5.2 Verify steam generator secondary side water levels are \geq 15% for required RCS loops.	12 hours
SR 3.4.5.3 Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

-----NOTES-----

1. All reactor coolant pumps (RCPs) and RHR pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:
 - a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
2. No RCP shall be started with any RCS cold leg temperature \leq the temperature below which LTOP is required as specified in the PTLR unless the pressurizer water level is less than 50%, OR the secondary side water temperature of each steam generator (SG) is $< 50^\circ\text{F}$ above each of the RCS cold leg temperatures.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required loop inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status.	Immediately
	<p><u>AND</u></p> <p>A.2 -----NOTE----- Only required if one RHR loop is OPERABLE. -----</p> <p>Be in MODE 5.</p>	

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two required loops inoperable. <u>OR</u> No RCS or RHR loop in operation.	B.1 Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	<u>AND</u> B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.6.1 Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2 Verify SG secondary side water levels are \geq 15% for required RCS loops.	12 hours
SR 3.4.6.3 Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

- a. One additional RHR loop shall be OPERABLE; or
- b. The secondary side water level of at least two steam generators (SGs) shall be $\geq 15\%$.

-----NOTES-----

1. The RHR pump of the loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided:
 - a. No operations are permitted that would cause reduction of the RCS boron concentration; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.
3. No reactor coolant pump shall be started with any RCS cold leg temperature \leq the temperature below which LTOP is required as specified in the PTLR unless the pressurizer water level is less than 50%, OR the secondary side water temperature of each SG is $< 50^\circ\text{F}$ above each of the RCS cold leg temperatures.
4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. <u>AND</u> Required SGs secondary side water levels not within limits.	A.1 Initiate action to restore a second RHR loop to OPERABLE status.	Immediately
	<u>OR</u> A.2 Initiate action to restore required SG secondary side water levels to within limits.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required RHR loops inoperable. <u>OR</u> No RHR loop in operation.	B.1 Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	<u>AND</u> B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.7.1 Verify one RHR loop is in operation.	12 hours
SR 3.4.7.2 Verify SG secondary side water level is \geq 15% in required SGs.	12 hours
SR 3.4.7.3 Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

-----NOTES-----

1. All RHR pumps may be removed from operation for ≤ 1 hour provided:
 - a. The core outlet temperature is maintained at least 10°F below saturation temperature.
 - b. No operations are permitted that would cause a reduction of the RCS boron concentration; and
 - c. No draining operations to further reduce the RCS water volume are permitted.
2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

-----NOTE-----

While this LCO is not met, entry into MODE 5, Loops Not Filled, from MODE 5, Loops Filled, is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required RHR loops inoperable. <u>OR</u> No RHR loop in operation.	B.1 Suspend all operations involving reduction in RCS boron concentration. <u>AND</u> B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.8.1 Verify one RHR loop is in operation.	12 hours
SR 3.4.8.2 Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

- LCO 3.4.9 The pressurizer shall be OPERABLE with:
- a. Pressurizer water level \leq 90%; and
 - b. Two groups of pressurizer heaters OPERABLE with the capacity of each group \geq 150 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3. <u>AND</u>	6 hours
	A.2 Fully insert all rods. <u>AND</u>	6 hours
	A.3 Place Rod Control System in a condition incapable of rod withdrawal. <u>AND</u>	6 hours
	A.4 Be in MODE 4.	12 hours
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. <u>AND</u>	6 hours
	C.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.9.1 Verify pressurizer water level is \leq 90%.	12 hours
SR 3.4.9.2 Verify capacity of each required group of pressurizer heaters is \geq 150 kW.	18 months
SR 3.4.9.3 Verify required pressurizer heaters are capable of being powered from an emergency power supply.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures $>$ the temperature below which LTOP is required as specified in the PTLR.

-----NOTE-----
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. <u>OR</u> Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with any RCS cold leg temperatures \leq the temperature below which LTOP is required as specified in the PTLR.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.10.1 Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within $\pm 1\%$.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

- NOTES-----
1. Separate Condition entry is allowed for each PORV.
 2. LCO 3.0.4 is not applicable.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable and not capable of being manually cycled.	B.1 Close associated block valve.	1 hour
	<u>AND</u>	
	B.2 Remove power from associated block valve.	1 hour
	<u>AND</u>	
	B.3 Restore the Class I PORV to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. One block valve inoperable.</p>	<p>C.1 Place associated PORV in manual control.</p>	<p>1 hour</p>
	<p><u>AND</u></p> <p>C.2 If the block valve is associated with a Class I PORV: Restore block valve to OPERABLE status.</p>	<p>72 hours</p>
	<p><u>OR</u></p> <p>C.3 If the block valve is associated with the non-Class I PORV: Close the block valve and remove its power.</p>	<p>72 hours</p>
<p>D. Required Action and associated Completion Time of Condition A, B, or C not met.</p>	<p>D.1 Initiate action to restore Class I PORV and/or associated block valves(s) to OPERABLE status.</p>	<p>Immediately</p>
	<p><u>AND</u></p> <p>D.2 Be in MODE 3.</p>	<p>6 hours</p>
	<p><u>AND</u></p> <p>D.3 Reduce Tavg to < 500°F.</p>	<p>12 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>E. Two Class I PORVs inoperable and not capable of being manually cycled.</p>	<p>E.1 Initiate action to restore Class I PORVs to OPERABLE status.</p> <p><u>AND</u></p>	<p>Immediately</p>
	<p>E.2 Close associated block valves</p> <p><u>AND</u></p>	<p>1 hour</p>
	<p>E.3 Remove power from associated block valves.</p> <p><u>AND</u></p>	<p>1 hour</p>
	<p>E.4 Be in MODE 3</p> <p><u>AND</u></p>	<p>6 hours</p>
	<p>E.5 Reduce Tavg to <500°F.</p>	<p>12 hours</p>
<p>F. More than one block valve inoperable.</p>	<p>F.1 Place associated PORVs in manual control.</p> <p><u>AND</u></p>	<p>1 hour</p>
	<p>F.2 Restore one block valve for a Class I PORV to OPERABLE status.</p> <p><u>AND</u></p>	<p>2 hours</p>
	<p>F.3 Restore remaining block valve for a Class I PORV to OPERABLE status.</p> <p><u>OR</u></p>	<p>72 hours</p>
	<p>F.4 If the remaining block valve is associated with the non-Class I PORV, close the block valve and remove its power.</p>	<p>72 hours</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time of Condition F not met.	G.1 Initiate action to restore block valve(s) to OPERABLE status.	Immediately
	<u>AND</u>	
	G.2 Be in MODE 3.	6 hours
	<u>AND</u>	
	G.3 Reduce Tavg to < 500°F.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.11.1 -----NOTE----- 1. Not required to be performed with block valve closed in accordance with the Required Action of Condition A, B, and E or Required Actions C.3 and F.4. 2. Only required to be performed in MODES 1 and 2. ----- Perform a complete cycle of each block valve.	92 days
-----NOTE----- Only required to be performed in MODES 1 and 2. -----	
SR 3.4.11.2 Perform a complete cycle of each PORV.	18 months
SR 3.4.11.3 Demonstrate OPERABILITY of the safety related nitrogen supply for the Class I PORVs.	18 months
SR 3.4.11.4 Not Used	

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with no safety injection pumps and a maximum of one centrifugal charging pump capable of injecting into the RCS and the accumulators isolated and one of the following pressure relief capabilities:

- a. Two Class I power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or
- b. The RCS depressurized and an RCS vent of ≥ 2.07 square inches.

-----NOTES-----

- 1. Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.
- 2. The limitation for a maximum of one charging pump capable of injecting into the RCS is not required for pump swap operation until 1 hour after completion of pump swap operation.

APPLICABILITY: MODE 4, when any RCS cold leg temperature is \leq the temperature below which LTOP is required as specified in the PTLR,
 MODE 5,
 MODE 6, when the reactor vessel head is on and the vessel head closure bolts are not fully de-tensioned.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more safety injection pumps capable of injecting into the RCS.	A.1 Initiate action to verify zero safety injection pumps are capable of injecting into the RCS.	Immediately
B. Two centrifugal charging pumps capable of injecting into the RCS.	B.1 Initiate action to verify a maximum of one centrifugal charging pump is capable of injecting into the RCS.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	C.1 Isolate affected accumulator.	1 hour
D. Required Action and associated Completion Time of Condition C not met.	D.1 Increase RCS cold leg temperature to > the temperature below which LTOP is required as specified in the PTLR. <u>OR</u> D.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours 12 hours
E. One required RCS Class I PORV inoperable in MODE 4.	E.1 Restore required RCS Class I PORV to OPERABLE status.	7 days
F. One required RCS Class I PORV inoperable in MODE 5 or 6, with the vessel head closure bolts not fully de-tensioned.	F.1 Restore required RCS Class I PORV to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>G. Two required RCS Class I PORVs inoperable.</p> <p><u>OR</u></p> <p>Required Action and associated Completion Time of Condition A, B, D, E, or F not met.</p> <p><u>OR</u></p> <p>LTOP System inoperable for any reason other than Condition A, B, C, D, E, or F.</p>	<p>G.1 Depressurize RCS and establish RCS vent of \geq 2.07 square inches.</p>	<p>8 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.12.1 Verify a maximum of zero safety injection pumps are capable of injecting into the RCS.</p>	<p>12 hours</p>
<p>SR 3.4.12.2 Verify a maximum of one centrifugal charging pump is capable of injecting into the RCS.</p>	<p>12 hours</p>
<p>SR 3.4.12.3 Verify each accumulator is isolated when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.</p>	<p>12 hours</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.4.12.4 Not used	
<p>SR 3.4.12.5 -----NOTE----- Only required to be performed when complying with LCO 3.4.12.b. -----</p> <p>Verify required RCS vent \geq 2.07 square inches open.</p>	<p>12 hours for unlocked open vent valve(s).</p> <p><u>AND</u></p> <p>31 days for vent valve(s) locked open, sealed, or otherwise secured in the open position.</p>
SR 3.4.12.6 Verify PORV block valve is open for each required Class I PORV.	72 hours
SR 3.4.12.7 Not used	
<p>SR 3.4.12.8 -----NOTE----- Not required to be performed until 12 hours after decreasing any RCS cold leg temperature to \leq the temperature below which LTOP is required as specified in the PTLR. -----</p> <p>Perform a COT on each required Class 1 PORV, excluding actuation.</p>	31 days
SR 3.4.12.9 Perform CHANNEL CALIBRATION for each required Class I PORV actuation channel.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

- a. No pressure boundary LEAKAGE;
- b. 1 gpm unidentified LEAKAGE;
- c. 10 gpm identified LEAKAGE;
- d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and
- e. 500 gallons per day primary to secondary LEAKAGE through any one SG.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.13.1 -----NOTE----- Not required to be performed until 12 hours after establishment of steady state operation. ----- Perform RCS water inventory balance.	72 hours
SR 3.4.13.2 Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the residual heat removal (RHR) flow path
when in, or during the transition to or from, the RHR mode
of operation.

ACTIONS

- NOTES-----
1. Separate Condition entry is allowed for each flow path.
 2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more flow paths with leakage from one or more RCS PIVs not within limit.</p>	<p>-----NOTE----- Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.1 and be in the reactor coolant pressure boundary or the high pressure portion of the system. -----</p>	<p>(continued)</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	<p><u>AND</u></p> A.2.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
	<p><u>OR</u></p> A.2.2 Restore RCS PIV to within limits.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.14.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. <p>-----</p> <p>Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.</p>	<p>In accordance with the Inservice Testing Program, and 18 months</p> <p><u>AND</u></p> <p>Within 24 hours following valve actuation due to automatic or manual action or flow through the valve except for valves 8802A, 8802B, and 8703</p>
<p>SR 3.4.14.2 Not used</p>	
<p>SR 3.4.14.3 Not used</p>	

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

- a. Both containment structure sumps and the reactor cavity sump level and flow monitor system,
- b. One containment atmosphere particulate radioactivity monitor and,
- c. Either a containment fan cooler unit (CFCU) condensate collection monitor or the containment atmosphere gaseous radioactivity monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS -----NOTE-----
LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment sump monitors inoperable.	<p>-----NOTE----- Not required until 12 hours after establishment of steady state operation. -----</p> <p>A.1 Perform SR 3.4.13.1.</p> <p><u>AND</u></p> <p>A.2 Restore required containment sump monitor to OPERABLE status.</p>	<p>Once per 24 hours</p> <p>30 days</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Required containment atmosphere particulate radioactivity monitor inoperable.</p>	<p>B.1.1 Analyze grab samples of the containment atmosphere.</p>	<p>Once per 24 hours</p>
	<p><u>OR</u></p> <p>B.1.2 -----NOTE----- Not required until 12 hours after establishment of steady state operation. -----</p>	
	<p>Perform SR 3.4.13.1.</p>	<p>Once per 24 hours</p>
	<p><u>AND</u></p> <p>B.2 Restore required containment atmosphere particulate radioactivity monitor to OPERABLE status.</p>	<p>30 days</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Required containment atmosphere gaseous radioactivity monitor inoperable.</p> <p><u>AND</u></p> <p>Required containment air cooler condensate flow rate monitor inoperable.</p>	<p>C.1.1 Analyze grab samples of the containment atmosphere</p>	<p>Once per 24 hours</p>
	<p><u>OR</u></p>	
	<p>C.1.2 -----NOTE----- Not required until 12 hours after establishment of steady state operation. -----</p>	
	<p>Perform SR 3.4.13.1</p>	<p>Once per 24 hours</p>
	<p><u>AND</u></p>	
	<p>C.2.1 Restore required containment atmosphere gaseous radioactivity monitor to OPERABLE status.</p>	<p>30 days</p>
<p><u>OR</u></p>		
<p>C.2.2 Restore required containment air cooler condensate flow rate monitor to OPERABLE status.</p>	<p>30 days</p>	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u> D.2 Be in MODE 5.	36 hours
E. All required monitors inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.15.1 Perform CHANNEL CHECK of the required containment atmosphere particulate and gaseous radioactivity monitors.	12 hours
SR 3.4.15.2 Perform CHANNEL FUNCTIONAL TEST of the required containment atmosphere particulate and gaseous radioactivity monitors.	92 days
SR 3.4.15.3 Perform CHANNEL CALIBRATION of the required containment sump monitors.	18 months
SR 3.4.15.4 Perform CHANNEL CALIBRATION of the required containment atmosphere particulate and gaseous radioactivity monitors.	18 months
SR 3.4.15.5 Perform CHANNEL CALIBRATION of the required CFCU condensate collection monitors.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) \geq 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 specific activity > 1.0 μ Ci/gm.	<p>----- LCO 3.0.4 is not applicable. -----</p> <p>A.1 Verify DOSE EQUIVALENT I-131 specific activity within the acceptable region of Figure 3.4.16-1.</p> <p><u>AND</u></p> <p>A.2 Restore DOSE EQUIVALENT I-131 specific activity to within limit.</p>	<p>Once per 4 hours</p> <p>48 hours</p>
B. Gross specific activity of the reactor coolant > 100/E μ Ci/gm.	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Reduce to $T_{avg} < 500^\circ\text{F}$.</p>	<p>6 hours</p> <p>12 hours</p>
<p>C. Required Action and associated Completion Time of Condition A not met.</p> <p><u>OR</u></p> <p>DOSE EQUIVALENT I-131 specific activity in the unacceptable region of Figure 3.4.16-1.</p>	<p>C.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>C.2 Reduce T_{avg} to $< 500^\circ\text{F}$.</p>	<p>6 hours</p> <p>12 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity $\leq 100/\bar{E}$ $\mu\text{Ci}/\text{gm}$.	7 days
SR 3.4.16.2	<p>-----NOTE----- Only required to be performed in MODE 1. ----- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 $\mu\text{Ci}/\text{gm}$.</p>	<p>14 days <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of $\geq 15\%$ RTP within a 1 hour period.</p>
SR 3.4.16.3	<p>-----NOTE----- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ----- Determine \bar{E} from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.</p>	184 days

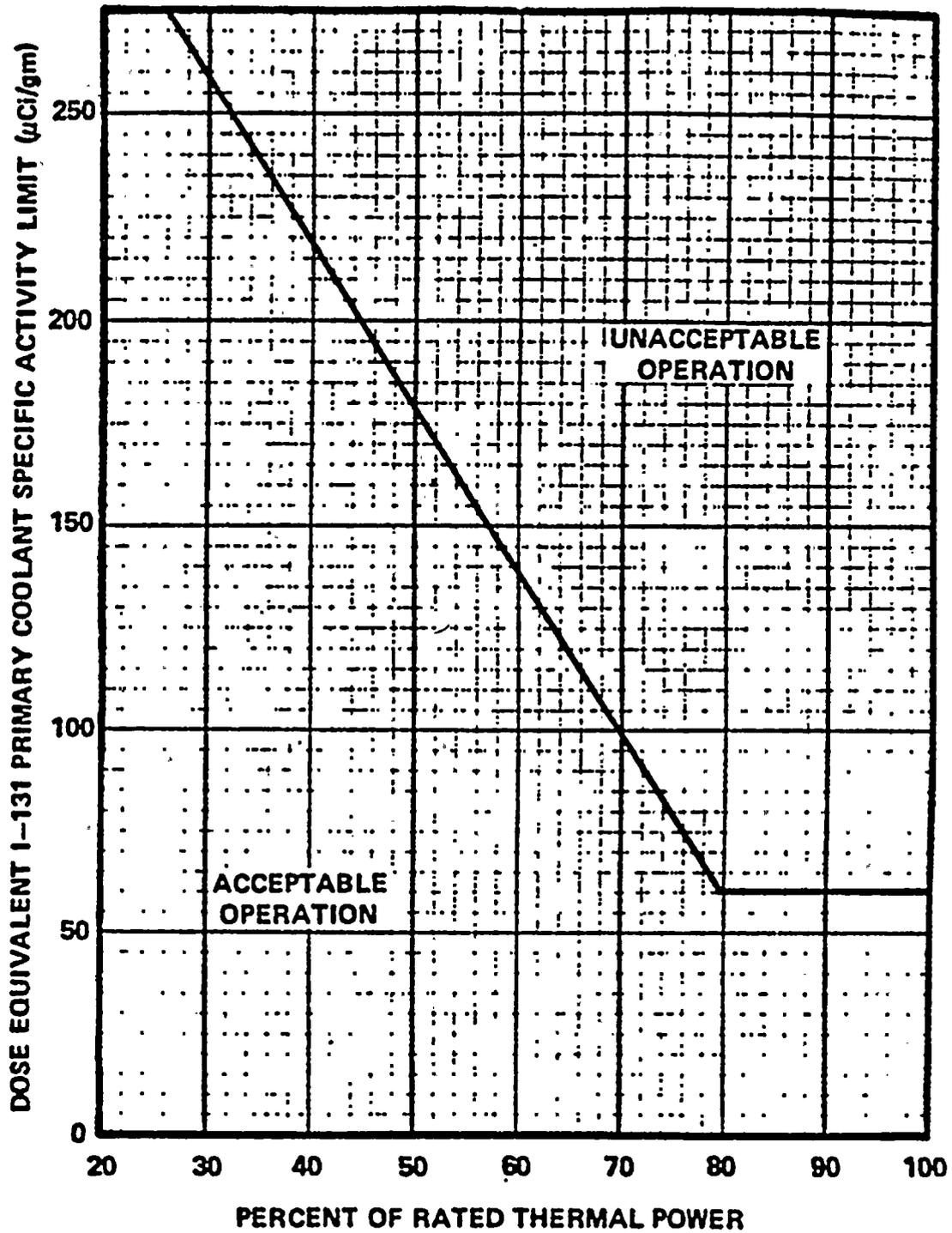


Figure 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT
VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT
SPECIFIC ACTIVITY > 1 µCI/GRAM DOSE EQUIVALENT I-131.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Four ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.
MODE 3 with RCS pressure > 1000 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One accumulator inoperable for reasons other than Condition A.	B.1 Restore accumulator to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to ≤1000 psig.	6 hours 12 hours
D. Two or more accumulators inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is $\geq 60.8\%$ and $\leq 72.6\%$.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 595.5 psig and ≤ 647.5 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2200 ppm and ≤ 2500 ppm.	31 days <u>AND</u> -----NOTE----- Only required to be performed for affected accumulators ----- Once within 6 hours after each solution volume increase of $\geq 5.6\%$ of narrow range indicated level that is not the result of addition from the refueling water storage tank.
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is >1000 psig.	31 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

-----NOTE-----

In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more trains inoperable.</p> <p><u>AND</u></p> <p>At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.</p>	<p>A.1 Restore train(s) to OPERABLE status.</p>	<p>72 hours</p>
<p>B. Required Action and associated Completion Time not met.</p>	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Be in MODE 4.</p>	<p>6 hours</p> <p>12 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY																																																
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed.	12 hours																																																
	<table border="1"> <thead> <tr> <th>Number</th> <th>Position</th> <th>Function</th> </tr> </thead> <tbody> <tr> <td>8703</td> <td>Closed</td> <td>RHR to RCS Hot Legs</td> </tr> <tr> <td>8802A</td> <td>Closed</td> <td>Safety Injection to RCS Hot Legs.</td> </tr> <tr> <td>8802B</td> <td>Closed</td> <td>Safety Injection to RCS Hot Legs</td> </tr> <tr> <td>8809A</td> <td>Open</td> <td>RHR to RCS Cold Legs</td> </tr> <tr> <td>8809B</td> <td>Open</td> <td>RHR to RCS Cold Legs</td> </tr> <tr> <td>8835</td> <td>Open</td> <td>Safety Injection to RCS Cold Legs</td> </tr> <tr> <td>8974A</td> <td>Open</td> <td>Safety Injection Pump Recir. to RWST</td> </tr> <tr> <td>8974B</td> <td>Open</td> <td>Safety Injection Pump Recir. to RWST</td> </tr> <tr> <td>8976</td> <td>Open</td> <td>RWST to Safety Injection Pumps</td> </tr> <tr> <td>8980</td> <td>Open</td> <td>RWST to RHR Pumps</td> </tr> <tr> <td>8982A</td> <td>Closed</td> <td>Containment Sump to RHR Pumps</td> </tr> <tr> <td>8982B</td> <td>Closed</td> <td>Containment Sump to RHR Pumps</td> </tr> <tr> <td>8992</td> <td>Open</td> <td>Spray Additive Tank to Educator</td> </tr> <tr> <td>8701</td> <td>Closed</td> <td>RHR Suction</td> </tr> <tr> <td>8702</td> <td>Closed</td> <td>RHR Suction</td> </tr> </tbody> </table>	Number	Position	Function	8703	Closed	RHR to RCS Hot Legs	8802A	Closed	Safety Injection to RCS Hot Legs.	8802B	Closed	Safety Injection to RCS Hot Legs	8809A	Open	RHR to RCS Cold Legs	8809B	Open	RHR to RCS Cold Legs	8835	Open	Safety Injection to RCS Cold Legs	8974A	Open	Safety Injection Pump Recir. to RWST	8974B	Open	Safety Injection Pump Recir. to RWST	8976	Open	RWST to Safety Injection Pumps	8980	Open	RWST to RHR Pumps	8982A	Closed	Containment Sump to RHR Pumps	8982B	Closed	Containment Sump to RHR Pumps	8992	Open	Spray Additive Tank to Educator	8701	Closed	RHR Suction	8702	Closed	RHR Suction	
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SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days																																																
SR 3.5.2.3	Verify ECCS piping is full of water.	31 days (continued)																																																

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY												
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program												
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months												
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months												
SR 3.5.2.7	Verify, for each ECCS throttle valve listed below, each mechanical position stop is in the correct position. <table border="0" style="margin-left: 40px;"> <tr> <td style="text-align: center;"><u>Charging Injection</u></td> <td style="text-align: center;"><u>Safety Injection</u></td> </tr> <tr> <td style="text-align: center;"><u>Throttle Valves</u></td> <td style="text-align: center;"><u>Throttle Valves</u></td> </tr> <tr> <td style="text-align: center;">8810A</td> <td style="text-align: center;">8822A</td> </tr> <tr> <td style="text-align: center;">8810B</td> <td style="text-align: center;">8822B</td> </tr> <tr> <td style="text-align: center;">8810C</td> <td style="text-align: center;">8822C</td> </tr> <tr> <td style="text-align: center;">8810D</td> <td style="text-align: center;">8822D</td> </tr> </table>	<u>Charging Injection</u>	<u>Safety Injection</u>	<u>Throttle Valves</u>	<u>Throttle Valves</u>	8810A	8822A	8810B	8822B	8810C	8822C	8810D	8822D	18 months
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<u>Throttle Valves</u>	<u>Throttle Valves</u>													
8810A	8822A													
8810B	8822B													
8810C	8822C													
8810D	8822D													
SR 3.5.2.8	Verify, by visual inspection, each ECCS train containment recirculation sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	18 months												

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS--Shutdown

LC0 3.5.3 One ECCS train shall be OPERABLE.

- - - - - NOTE - - - - -
 An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation.
 - - - - -

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required ECCS residual heat removal (RHR) subsystem inoperable.	A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status	Immediately
B. Required ECCS Centrifugal Charging Pump subsystem inoperable.	B.1 Restore required ECCS Centrifugal Charging Pump subsystem to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 5.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	The following SRs are applicable for all equipment required to be OPERABLE: SR 3.5.2.1 SR 3.5.2.7 SR 3.5.2.3 SR 3.5.2.8 SR 3.5.2.4	In accordance with applicable SRs

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. RWST boron concentration not within limits.</p> <p><u>OR</u></p> <p>RWST borated water temperature not within limits.</p>	<p>A.1 Restore RWST to OPERABLE status.</p>	<p>8 hours</p>
<p>B. RWST inoperable for reasons other than Condition A.</p>	<p>B.1 Restore RWST to OPERABLE status.</p>	<p>1 hour</p>
<p>C. Required Action and associated Completion Time not met.</p>	<p>C.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>C.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	<p>-----NOTE----- Only required to be performed when ambient air temperature is < 35°F. -----</p> <p>Verify RWST borated water temperature is ≥ 35°F.</p>	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 400,000 gallons (81.5% indicated level.)	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2300 ppm and ≤ 2500 ppm.	7 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow shall be ≤ 40 gpm with RCS pressure ≥ 2215 psig and ≤ 2255 psig and the charging flow control valve full open.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Seal injection flow not within limit.	A.1 Verify $\geq 100\%$ flow equivalent to a single OPERABLE ECCS charging train is available <u>AND</u> A.2 Adjust manual seal injection throttle valves to give a flow within limit with RCS pressure ≥ 2215 psig and ≤ 2255 psig and the charging flow control valve full open.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.5.5.1 -----NOTE----- Not required to be performed until 4 hours after the Reactor Coolant System pressure stabilizes at ≥ 2215 psig and ≤ 2255 psig. -----</p> <p>Verify manual seal injection throttle valves are adjusted to give a flow within limit with RCS pressure ≥ 2215 psig and ≤ 2255 psig and the charging flow control valve full open.</p>	<p>31 days</p>

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.1 Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program.
SR 3.6.1.2 Not used	

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

- NOTES-----
1. Entry and exit is permissible to perform repairs on the affected air lock components.
 2. Separate Condition entry is allowed for each air lock.
 3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more containment air locks with one containment air lock door inoperable.</p>	<p>-----NOTES-----</p> <ol style="list-style-type: none"> 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. <p>-----</p>	<p>(continued)</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Verify the OPERABLE door is closed in the affected air lock.	1 hour
	<u>AND</u>	
	A.2 Lock the OPERABLE door closed in the affected air lock.	24 hours
	<u>AND</u>	
	A.3 -----NOTE----- Air lock doors in high radiation areas may be verified locked closed by administrative means. ----- Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. One or more containment air locks with containment air lock interlock mechanism inoperable.</p>	<p>-----NOTES----- 1.Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2.Entry and exit of containment is permissible under the control of a dedicated individual. -----</p>	
	<p>B.1 Verify an OPERABLE door is closed in the affected air lock.</p>	<p>1 hour</p>
	<p><u>AND</u></p>	
	<p>B.2 Lock an OPERABLE door closed in the affected air lock.</p>	<p>24 hours</p>
<p><u>AND</u></p>		
<p>B.3 -----NOTE----- Air lock doors in high radiation areas may be verified locked closed by administrative means. -----</p> <p>Verify an OPERABLE door is locked closed in the affected air lock.</p>	<p>Once per 31 days</p>	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. One or more containment air locks inoperable for reasons other than Condition A or B.</p>	<p>C.1 Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.</p> <p><u>AND</u></p> <p>C.2 Verify a door is closed in the affected air lock.</p> <p><u>AND</u></p> <p>C.3 Restore air lock to OPERABLE status.</p>	<p>Immediately</p> <p>1 hour</p> <p>24 hours</p>
<p>D. Required Action and associated Completion Time not met.</p>	<p>D.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>D.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.2.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1 in accordance with the Containment Leakage Rate Testing Program. <p>-----</p> <p>Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.</p>	<p>In accordance with the Containment Leakage Rate Testing Program</p>
<p>SR 3.6.2.2 Verify only one door in the air lock can be opened at a time.</p>	<p>24 months</p>

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

-----NOTE-----
 Not applicable to Main Steam Safety Valves (MSSVs), Main Steam Isolation Valves (MSIVs), Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves, and Atmospheric Dump Valves (ADV).

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

- NOTES-----
1. Penetration flow path(s) except no more than two of three flow paths for containment purge supply and exhaust and containment vacuum/pressure relief paths at one time may be unisolated intermittently under administrative controls.
 2. Separate Condition entry is allowed for each penetration flow path.
 3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.
 4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.
-

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----NOTE----- Only applicable to penetration flow paths with two containment isolation valves. ----- One or more penetration flow paths with one containment isolation valve inoperable except for a containment purge supply and exhaust valve or pressure/vacuum relief valve leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.2 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. <p>-----</p> <p>Verify the affected penetration flow path is isolated.</p>	<p>Once per 31 days for isolation devices outside containment</p> <p><u>AND</u></p> <p>Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. One or more penetration flow paths with one or more containment purge supply and exhaust and vacuum/pressure relief valves not within purge valve leakage limits.</p>	<p>D.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.</p> <p><u>AND</u></p> <p>D.2 -----NOTES----- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. -----</p> <p>Verify the affected penetration flow path is isolated.</p> <p><u>AND</u></p> <p>D.3 Perform SR 3.6.3.7 for the resilient seal purge or vacuum/pressure relief valves closed to comply with Required Action D.1.</p>	<p>24 hours</p> <p>Once per 31 days for isolation devices outside containment</p> <p><u>AND</u></p> <p>Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment</p> <p>Once per 92 day</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours
	<u>AND</u> E.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.3.1 Not used	
SR 3.6.3.2 Verify each 48 inch containment purge supply and exhaust and 12 inch vacuum/pressure relief valve is closed, except when these valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	31 days
SR 3.6.3.3 -----NOTE----- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ----- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.3.4 -----NOTE----- Valves and blind flanges in high radiation areas may be verified by use of administrative means. -----</p> <p>Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.</p>	<p>Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days</p>
<p>SR 3.6.3.5 Verify the isolation time of each automatic power operated containment isolation valve that is not locked, sealed or otherwise secured in position is within limits.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.6.3.6 Not used</p>	

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.3.7 -----NOTE----- This surveillance is not required when the penetration flow path is isolated by a leak tested blank flange. ----- Perform leakage rate testing for containment purge supply and exhaust and vacuum/pressure relief valves with resilient seals in accordance with the Containment Leakage Rate Testing Program.</p>	<p>184 days <u>AND</u> Within 92 days after opening the valve</p>
<p>SR 3.6.3.8 Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.</p>	<p>18 months</p>
<p>SR 3.6.3.9 Not used</p>	
<p>SR 3.6.3.10 Verify each 12 inch containment vacuum/pressure relief valve is blocked to restrict the valve from opening > 50°.</p>	<p>18 months</p>
<p>SR 3.6.3.11 Not used</p>	

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -1.0 psig and $\leq +1.2$ psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.4.1 Verify containment pressure is within limits.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LC0 3.6.5 Containment average air temperature shall be $\leq 120^{\circ}\text{F}$.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.5.1 Verify containment average air temperature is within limit.	24 hours

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment fan cooling unit (CFCU) trains with either:

- a. Four CFCUs, or
- b. Three CFCUs, each of the three supplied from a different vital bus.

shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 84 hours
C. One required CFCU train inoperable such that a minimum of two CFCUs remain OPERABLE.	C.1 Restore required CFCU train to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>D. One required containment spray train inoperable and one required CFCU train inoperable such that a minimum of two CFCUs remain OPERABLE.</p>	<p>D.1 Restore one required containment spray train to OPERABLE status, <u>OR</u> D.2 Restore one CFCU train to OPERABLE status such that four CFCUs or three CFCUs, each supplied by a different vital bus, are OPERABLE.</p>	<p>72 hours 72 hours</p>
<p>E. Required Action and associated Completion Time of Condition C or D not met.</p>	<p>E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 5.</p>	<p>6 hours 36 hours</p>
<p>F. Two containment spray trains inoperable. <u>OR</u> One containment spray train inoperable and two CFCU trains inoperable such that one or less CFCUs remain OPERABLE. <u>OR</u> One or less CFCUs OPERABLE</p>	<p>F.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.2	Operate each required CFCU for ≥ 15 minutes.	31 days
SR 3.6.6.3	Verify component cooling water flow rate to each required CFCU is ≥ 1650 gpm.	31 days
SR 3.6.6.4	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.5	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.6.6.6	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.6.7	Verify each CFCU starts automatically on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.6.8 Verify each spray nozzle is unobstructed.	10 years
SR 3.6.6.9 Verify each CFCU starts on low speed.	31 days

3.6 CONTAINMENT SYSTEMS

3.6.7 Spray Additive System

LCO 3.6.7 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.7.1 Verify each spray additive manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.7.2	Verify spray additive tank solution volume is \geq 46.2% and \leq 91.9%.	184 days
SR 3.6.7.3	Verify spray additive tank NaOH solution concentration is \geq 30% and \leq 32% by weight.	184 days
SR 3.6.7.4	Verify each spray additive automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.6.7.5	Verify spray additive flow from each solution's flow path.	5 years

3.6 CONTAINMENT SYSTEMS

3.6.8 Hydrogen Recombiners

LCO 3.6.8 Two hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One hydrogen recombinder inoperable.	A.1 -----NOTE----- LCO 3.0.4 is not applicable. ----- Restore hydrogen recombinder to OPERABLE status.	30 days
B. Two hydrogen recombiners inoperable.	B.1 Verify by administrative means that the hydrogen control function is maintained. <u>AND</u> B.2 Restore one hydrogen recombinder to OPERABLE status.	1 hour <u>AND</u> Once per 12 hours thereafter 7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Perform a system functional test for each hydrogen recombiner.	18 months
SR 3.6.8.2	Visually examine each hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	18 months
SR 3.6.8.3	Perform a resistance to ground test for each heater phase.	18 months

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 Five MSSVs per steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more MSSVs inoperable.</p>	<p>A.1 Reduce Thermal Power to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.</p>	<p>4 hours</p>
	<p><u>AND</u></p> <p>A.2 -----NOTE----- Only applicable in MODE 1. ----- Reduce the Power Range High Neutron Flux trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.</p>	<p>72 hours</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Required Action and associated Completion Time not met.</p> <p><u>OR</u></p> <p>One or more steam generators with less than two MSSVs OPERABLE.</p>	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Be in MODE 4.</p>	<p>6 hours</p> <p>12 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.1.1</p> <p>-----NOTE----- Only required to be performed in MODES 1 and 2. -----</p> <p>Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within $\pm 1\%$.</p>	<p>In accordance with the Inservice Testing Program</p>

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE (%RTP)
4	92
3	45
2	27

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER				LIFTSETTING (psig ± 3 %) -----NOTE----- Lowest set MSSV Lift Setting Tolerance is +3/-2% -----
#1	STEAM GENERATOR		#4	
	#2	#3		
RV-3	RV-7	RV-11	RV-58	1065
RV-4	RV-8	RV-12	RV-59	1078
RV-5	RV-9	RV-13	RV-60	1090
RV-6	RV-10	RV-14	RV-61	1103
RV-222	RV-223	RV-224	RV-225	1115

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Four MSIVs shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 except when all MSIVs are closed and de-activated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MSIV inoperable in MODE 1.	A.1 Restore MSIV to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
C. -----NOTE----- Separate Condition entry is allowed for each MSIV. ----- One or more MSIVs inoperable in MODE 2 or 3.	C.1 Close MSIV. <u>AND</u> C.2 Verify MSIV is closed.	8 hours Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE	FREQUENCY
<p>SR 3.7.2.1</p> <p>-----NOTE----- Only required to be performed in MODES 1 and 2.</p> <p>Verify closure time of each MSIV is \leq 5 seconds.</p>	<p>In accordance with the Inservice Testing Program</p>
<p>SR 3.7.2.2</p> <p>-----NOTE----- Only required to be performed in MODES 1 and 2.</p> <p>Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.</p>	<p>18 months</p>

3.7 PLANT SYSTEMS

3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves

LC0 3.7.3 Four MFIVs, four MFRVs, and associated bypass valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when MFIV, MFRV, or associated bypass valve is closed and de-activated or isolated by a closed manual valve.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFIVs inoperable.	A.1 Close or isolate MFIV.	72 hours
	<u>AND</u> A.2 Verify MFIV is closed or isolated.	Once per 7 days
B. One or more MFRVs inoperable.	B.1 Close or isolate MFRV.	72 hours
	<u>AND</u> B.2 Verify MFRV is closed or isolated.	Once per 7 days
C. One or more MFRV bypass valves inoperable.	C.1 Close or isolate bypass valve.	72 hours
	<u>AND</u> C.2 Verify bypass valve is closed or isolated.	Once per 7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two valves in the same flow path inoperable.	D.1 Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3.	6 hours
	<u>AND</u> E.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFIV is \leq 60 seconds.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify the closure time of each MFRV and associated bypass valve is \leq 7 seconds.	At each COLD SHUTDOWN, but not more frequently than once per 92 days.
SR 3.7.3.3	Verify each MFIV actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.4 10% Atmospheric Dump Valves (ADVs)

LCO 3.7.4 Four ADV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ADV line inoperable.	A.1 -----NOTE----- LCO 3.0.4 is not applicable. ----- Restore required ADV line to OPERABLE status.	7 days
B. Two required ADV lines inoperable.	B.1 Restore all but one ADV line to OPERABLE status.	72 hours
C. Three or more required ADV lines inoperable.	C.1 Restore all but two ADV lines to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each ADV.	18 months
SR 3.7.4.2	Verify one complete cycle of each ADV block valve.	18 months
SR 3.7.4.3	Verify that the backup air bottle for each ADV has a pressure ≥ 260 psig.	24 hours

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

-----NOTE-----
Only one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO
B. One AFW train inoperable in MODE 1, 2 or 3 for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Required Action and associated Completion Time for Condition A or B not met.</p> <p><u>OR</u></p> <p>Two AFW trains inoperable in MODE 1, 2, or 3.</p>	<p>C.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>C.2 Be in MODE 4.</p>	<p>6 hours</p> <p>18 hours</p>
<p>D. Three AFW trains inoperable in MODE 1, 2, or 3.</p>	<p>D.1</p> <p>-----NOTE----- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. -----</p> <p>Initiate action to restore one AFW train to OPERABLE status.</p>	<p>Immediately</p>
<p>E. Required AFW train inoperable in MODE 4.</p>	<p>E.1 Initiate action to restore AFW train to OPERABLE status.</p>	<p>Immediately</p>
<p>F. With the CST flow path not open to the AFW pump suction.</p>	<p>F.1 Restore the CST flow path.</p>	<p>4 hours</p>
<p>G. With the FWST flow path not capable of being aligned to the AFW pump suction.</p>	<p>G.1 Restore the capability of FWST alignment.</p>	<p>7 days</p>

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Required Action and associated Completion Time for Condition F or G not met	H.1 Be in MODE 3	6 hours
	<u>AND</u> H.2 Be in MODE 4 without reliance upon steam generator for heat removal.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.5.1 Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2 -----NOTE----- Not required to be performed for the turbine driven AFW pump until 24 hours after \geq 650 psig in the steam generator. ----- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Test Program.
SR 3.7.5.3 -----NOTE----- Not applicable in MODE 4 when steam generator is relied upon for heat removal. ----- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.7.5.4</p> <p style="text-align: center;">-----NOTES-----</p> <p>1. Not required to be performed for the turbine driven AFW pump until 24 hours after \geq 650 psig in the steam generator.</p> <p>2. Not applicable in MODE 4 when generator is relied upon for heat removal.</p> <p style="text-align: center;">-----</p> <p>Verify each AFW pump starts automatically on an actual or simulated actuation signal.</p>	<p>18 months</p>
<p>SR 3.7.5.5 Not Used.</p>	
<p>SR 3.7.5.6 Verify the FWST is capable of being aligned to the AFW system by cycling each FWST valve in the flow path necessary for realignment through at least one full cycle.</p>	<p>92 days</p>

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tank (CST) and Fire Water Storage Tank (FWST)

LCO 3.7.6 The CST level shall be $\geq 41.3\%$ and the FWST level shall be $\geq 22.2\%$ for one unit operation and $\geq 41.7\%$ for two unit operation.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST level not within limit.	A.1 Restore CST level to within limit.	4 hours
B. FWST level not within limit.	B.1 Restore FWST level to within limit.	7 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 4, without reliance on steam generator for heat removal.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify the CST level is ≥ 41.3 %.	12 hours
SR 3.7.6.2	Verify the FWST level is ≥ 22.2 % for one unit operation and ≥ 41.7 % for two unit operation.	12 hours

3.7 PLANT SYSTEMS

3.7.7 Vital Component Cooling Water (CCW) System

LCO 3.7.7 Two vital CCW loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One vital CCW loop inoperable.	<p>A.1 -----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW. -----</p> <p>Restore vital CCW loop to OPERABLE status.</p>	72 hours
B. Required Action and associated Completion Time of Condition A not met.	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.7.7.1 -----NOTE----- Isolation of CCW flow to individual components does not render the CCW System inoperable -----</p> <p>Verify each vital CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.</p>	<p>31 days</p>
<p>SR 3.7.7.2 Verify each vital CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.</p>	<p>18 months</p>
<p>SR 3.7.7.3 Verify each vital CCW pump starts automatically on an actual or simulated actuation signal.</p>	<p>18 months</p>

3.7 PLANT SYSTEMS

3.7.8 Auxiliary Saltwater System (ASW)

LCO 3.7.8 Two ASW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One ASW train inoperable.</p>	<p>A.1 -----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by ASW. ----- Restore ASW train to OPERABLE status.</p>	<p>72 hours</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p>	<p>B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.</p>	<p>6 hours 36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.8.1 Verify each ASW manual and power operated, valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position or that a motive force is available such that the valve would be capable of being placed in the correct position.	31 days
SR 3.7.8.2 Verify each ASW power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, can be moved to the correct position.	In accordance with the Inservice Test Program.
SR 3.7.8.3 Verify each ASW pump starts automatically on an actual or simulated actuation signal.	18 months

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. With the UHS inoperable.	A.1 Place a second CCW heat exchanger in service.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Not Used	
SR 3.7.9.2	Verify water temperature of UHS is within limits.	24 hours if UHS temperature is $\leq 60^{\circ}\text{F}$. <u>AND</u> 12 hours if UHS temperature $> 60^{\circ}\text{F}$ but $\leq 62^{\circ}\text{F}$. <u>AND</u> 2 hours if UHS temperature $> 62^{\circ}\text{F}$ but $\leq 64^{\circ}\text{F}$.

3.7 PLANT SYSTEMS

3.7.10 Control Room Ventilation System (CRVS)

LCO 3.7.10 Two CRVS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRVS train inoperable.	A.1 Restore CRVS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies	C.1 Place OPERABLE CRVS train in recirculation mode. <u>OR</u>	Immediately
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. (continued)</p>	<p>C.2.1 Suspend CORE ALTERATIONS.</p>	<p>Immediately</p>
	<p><u>AND</u> C.2.2 Suspend movement of irradiated fuel assemblies.</p>	<p>Immediately</p>
<p>D. Two CRVS trains inoperable for reasons other than Condition D in MODE 5 or 6, or during movement of irradiated fuel assemblies</p>	<p>D.1 Suspend CORE ALTERATIONS. <u>AND</u> D.2 Suspend movement of irradiated fuel assemblies.</p>	<p>Immediately Immediately</p>
<p>E. Two CRVS trains inoperable for reasons other than Condition D in MODE 1, 2, 3, or 4.</p>	<p>E.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRVS train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.10.2	Perform required CRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each CRVS train automatically switches into the pressurization mode of operation on an actual or simulated actuation signal.	18 months
SR 3.7.10.4	Verify one CRVS train can maintain a positive pressure of ≥ 0.125 inches water gauge, relative to the outside atmosphere during the pressurization mode of operation.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

LCO 3.7.11 Not Used

3.7 PLANT SYSTEMS

3.7.12 Auxiliary Building Ventilation System (ABVS)

LCO 3.7.12 Two ABVS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The common HEPA filter and/or charcoal adsorber inoperable.	A.1 Restore the common HEPA filter and charcoal adsorber to OPERABLE status.	24 hours
B. One ABVS train inoperable.	B.1 Restore ABVS train to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met..	C.1 Be in MODE 3.	6 hours
	<u>AND</u> C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.12.1 Operate each ABVS train for \geq 15 minutes, and one train for \geq 10 continuous hours with the heater operating .	31 days
SR 3.7.12.2 Perform required ABVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.12.3	Verify each ABVS train actuates on an actual or simulated actuation signal and the system realigns to exhaust through the common HEPA filter and charcoal adsorber.	18 months
SR 3.7.12.4	Not Used.	
SR 3.7.12.5	Not Used.	
SR 3.7.12.6	Verifying that leakage through the ABVS Dampers M2A and M2B is less than or equal to 5 cfm when subjected to a Constant Pressure or Pressure Decay Leak Rate Test in accordance with ASME N510-1989. The test pressure for the leak rate test shall be based on a maximum operating pressure as defined in ASME N510-1989, of 8 inches water gauge.	18 months

3.7 PLANT SYSTEMS

3.7.13 Fuel Handling Building Ventilation System (FHBVS)

LCO 3.7.13 Two FHBVS trains shall be OPERABLE.

APPLICABILITY: During movement of irradiated fuel assemblies in the fuel handling building.

-----NOTE-----
LCO 3.0.3 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One FHBVS train inoperable.	A.1 Restore FHBVS train to OPERABLE status.	Immediately
	<u>OR</u>	
	A.2 Place the OPERABLE FHBV train in operation.	Immediately
	<u>OR</u>	
	A.3 Suspend movement of irradiated fuel assemblies in the fuel handling building.	Immediately
B. Not Used		
C. Not Used		
D. Two FHBVS trains inoperable during movement of irradiated fuel assemblies in the fuel building.	D.1 Suspend movement of irradiated fuel assemblies in the fuel handling building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each FHBVS train for \geq 15 minutes.	31 days
SR 3.7.13.2	Perform required FHBVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each FHBVS train actuates on an actual or simulated actuation signal.	18 months
SR 3.7.13.4	Verify one FHBVS train can maintain a pressure \leq -0.125 inches water gauge with respect to atmospheric pressure during the post accident mode of operation.	18 months on a STAGGERED TEST BASIS

3.7 PLANT SYSTEMS

3.7.14 Penetration Room Exhaust Air Cleanup System (PREACS)

LC0 3.7.14 Not Used

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Pool Water Level

LCO 3.7.15 The spent fuel pool water level shall be \geq 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool water level not within limit.	A.1 -----NOTE----- LCO 3.0.3 is not applicable. ----- Suspend movement of irradiated fuel assemblies in the spent fuel pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.15.1 Verify the spent fuel pool water level is \geq 23 ft above the top of the irradiated fuel assemblies seated in the storage racks.	7 days

3.7 PLANT SYSTEMS

3.7.16 Spent Fuel Pool Boron Concentration

LCO 3.7.16 The spent fuel pool boron concentration shall be \geq 2000 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	-----NOTE----- LCO 3.0.3 is not applicable. -----	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	<u>AND</u> A.2 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.16.1 Verify the spent fuel pool boron concentration is within limit.	31 days

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17.1 The combination of initial enrichment, initial B-10 content, burnup, and storage pattern of each spent fuel assembly stored in Region 1 shall be:

- a. The initial enrichment is 4.5 weight percent U-235 or less; or
- b. The initial enrichment is from 4.5 up to a maximum of 5.0 weight percent U-235, and any of the following conditions are met:
 - 1) The combination of initial enrichment and cumulative burnup of the assemblies is within the acceptable area of Figure 3.7.17-1; or
 - 2) The assemblies initially contained a minimum of a nominal 36 mg/in. per assembly of the isotope B-10 integrated in the fuel rods; or
 - 3) The assemblies are put in a checkerboard pattern with any of the following:
 - a) water cells, or
 - b) assemblies that initially contained a minimum of a nominal 72 mg/in. per assembly of the isotope B-10 integrated in the fuel rods, or
 - c) partially irradiated fuel of at least 8000 MWD/MTU cumulative burnup; or
 - 4) The assemblies are put into a pattern with alternate rows of fuel assemblies and water cells.

APPLICABILITY: Whenever any fuel assembly is stored in Region 1 of the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 -----NOTE----- LCO 3.0.3 is not applicable. ----- Initiate action to move the noncomplying fuel assembly into an acceptable pattern that complies with this LCO or LCO 3.7.17.2.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.17.1.1 Verify by administrative means that the fuel assembly characteristics and its expected location is in accordance with LCO 3.7.17.1.	Prior to each fuel assembly move, when the assembly will be stored in Region 1.

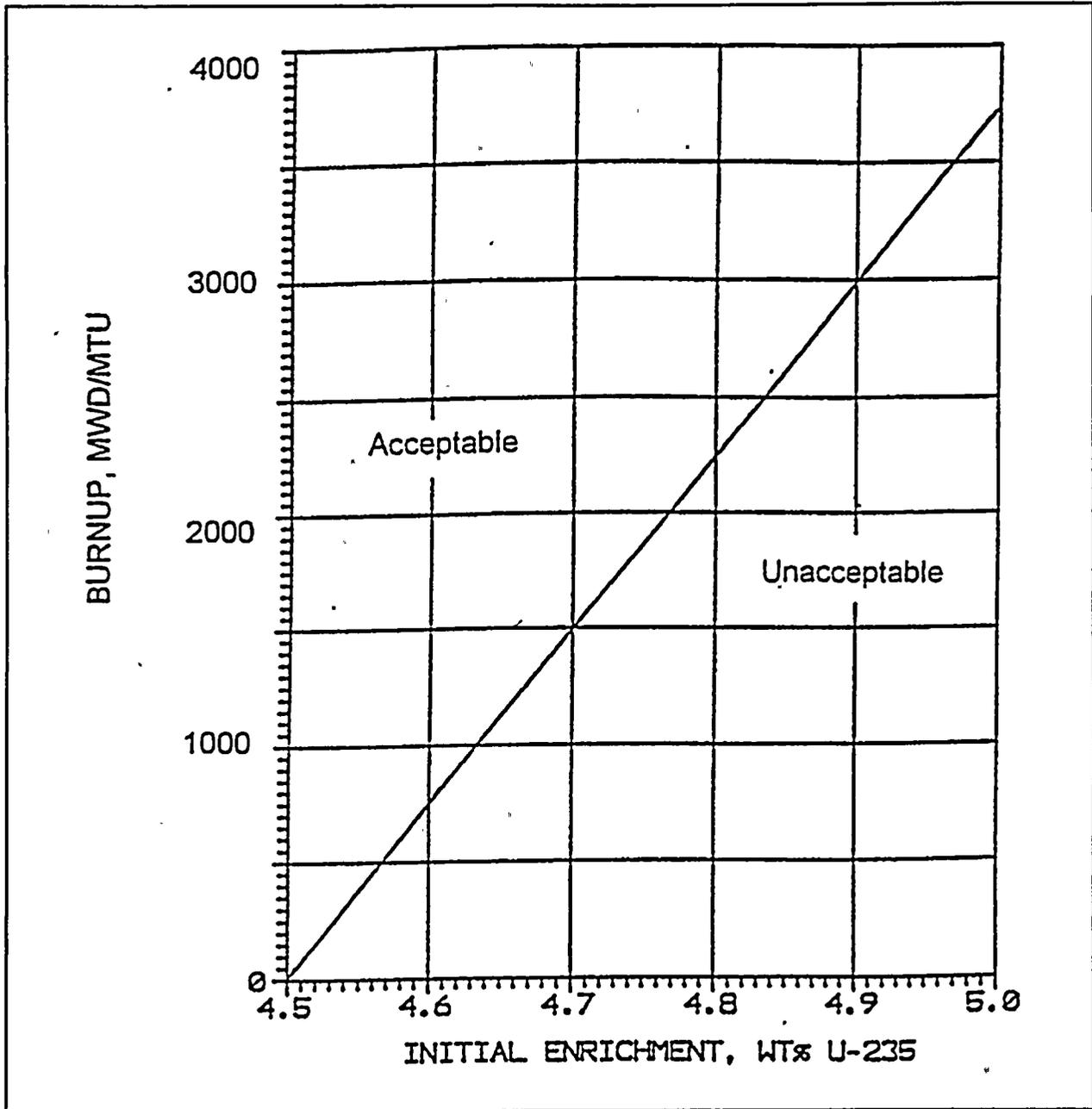


FIGURE 3.7.17-1
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT (NO IFBA) TO PERMIT
STORAGE IN REGION 1

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17.2 The combination of initial enrichment, fuel pellet diameter and burnup of each spent fuel assembly stored in Region 2 shall be within the acceptable area of Figure 3.7.17-2, or the fuel assembly is stored in a checker board pattern with water cells or non-fissile material.

APPLICABILITY: Whenever any fuel assembly is stored in Region 2 of the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	<p>A.1</p> <p>-----NOTE----- LCO 3.0.3 is not applicable. -----</p> <p>Initiate action to move the noncomplying fuel assembly into an acceptable pattern that complies with this LCO or LCO 3.7.17.1.</p>	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.17.2.1 Verify by administrative means that the fuel assembly characteristics and its expected storage location is in accordance with LCO 3.7.17.2.	Prior to each fuel assembly move, when the assembly will be stored in Region 2.

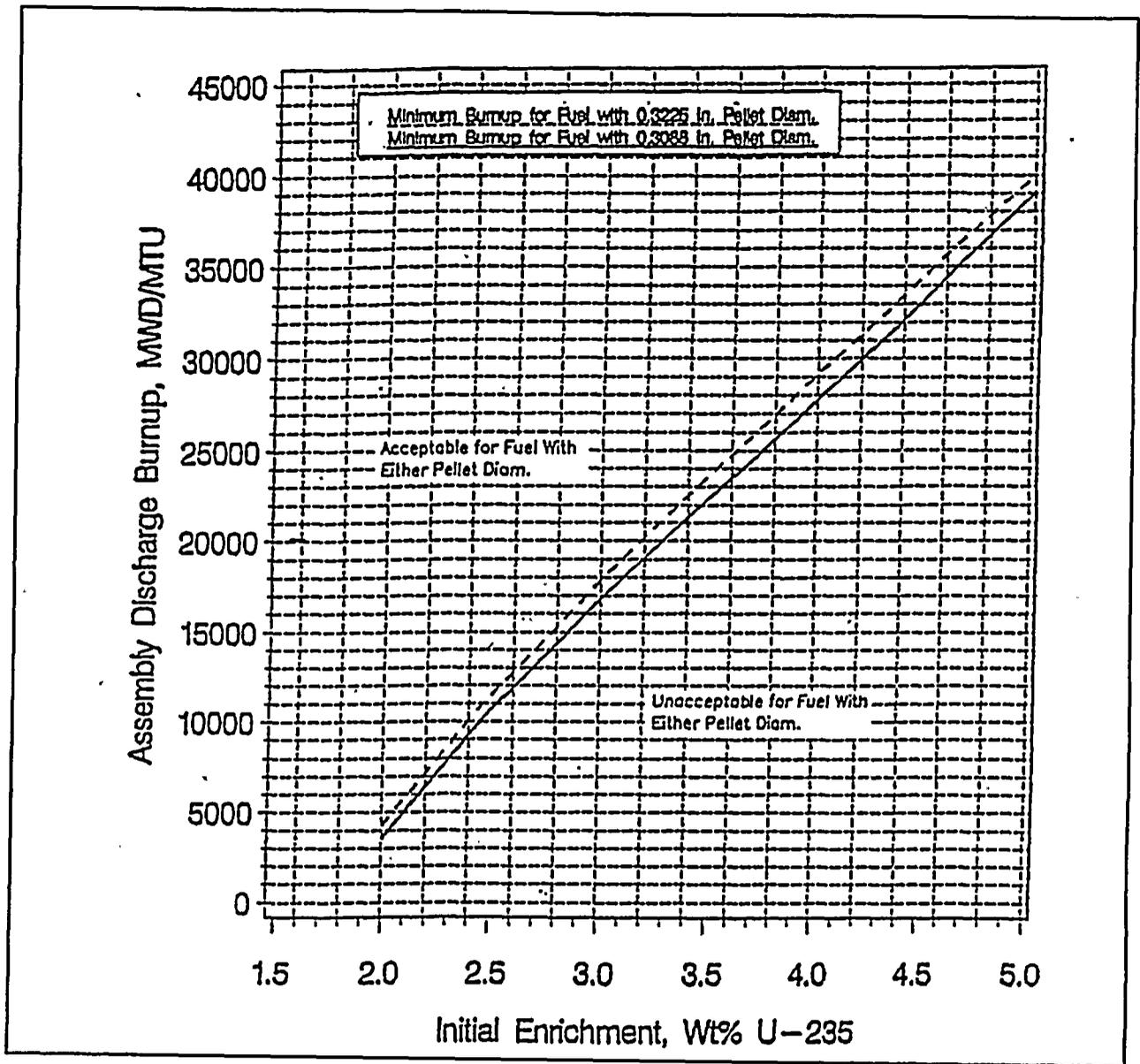


FIGURE 3.7.17-2
MINIMUM REQUIRED ASSEMBLY DISCHARGE BURNUP
AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT
STORAGE IN REGION 2

3.7 PLANT SYSTEMS

3.7.18 Secondary Specific Activity

LCO 3.7.18 The specific activity of the secondary coolant shall be $\leq 0.10 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3.	6 hours
	<u>AND</u> A.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.18.1 Verify the specific activity of the secondary coolant is $\leq 0.10 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources – Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

- a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and
- b. Three diesel generators (DGs) capable of supplying the onsite Class 1E power distribution subsystem(s); and
- c. Two supply trains of the diesel fuel oil (DFO) transfer system.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for required OPERABLE offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> A.2 Restore required offsite circuit to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. One DG inoperable.</p>	<p>B.1 Perform SR 3.8.1.1 for the required offsite circuit(s).</p>	<p>1 hour <u>AND</u> Once per 8 hours thereafter</p>
	<p><u>AND</u></p> <p>-----NOTE----- In MODE 1, 2, and 3, TDAFW pump is considered a required redundant feature. -----</p>	
	<p>B.2 Declare required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable.</p>	<p>4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)</p>
	<p><u>AND</u></p> <p>B.3.1 Determine OPERABLE DG(s) is not inoperable due to common cause failure.</p> <p><u>OR</u></p>	<p>24 hours</p>
	<p>B.3.2 Perform SR 3.8.1.2 for OPERABLE DG(s).</p> <p><u>AND</u></p>	<p>24 hours</p> <p>(continued)</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Restore DG to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. <u>AND</u> C.2 Restore one required offsite circuit to OPERABLE status.	12 hours from discovery of Condition C concurrent with inoperability of redundant required features 24 hours
D. One required offsite circuit inoperable. <u>AND</u> One DG inoperable.	D.1 Restore required offsite circuit to OPERABLE status. <u>OR</u> D.2 Restore DG to OPERABLE status.	12 hours 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two or more DGs inoperable.	E.1 Ensure at least two DGs are OPERABLE.	2 hours
F. One supply train of the DFO transfer system inoperable.	F.1 Restore the DFO transfer system to OPERABLE status.	72 hours
G. Two supply trains of the DFO transfer system inoperable.	G.1 Restore one train of the DFO transfer system to OPERABLE status.	1 hour
H. Required Action and associated Completion Time of Condition A, B, C, D, E, F or G not met.	H.1 Be in MODE 3. <u>AND</u> H.2 Be in MODE 5.	6 hours 36 hours
I. Two or more DGs inoperable. <u>AND</u> One or more required offsite circuits inoperable.	I.1 Enter LCO 3.0.3.	Immediately
J. One or more DGs inoperable. <u>AND</u> Two required offsite circuits inoperable.	J.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.1.1 Verify correct breaker alignment and indicated power availability for each required offsite circuit.	7 days
SR 3.8.1.2 -----NOTES----- 1. Performance of SR 3.8.1.7 satisfies this SR. 2. All DG starts may be preceded by an engine prelube period and followed by a warmup period prior to loading. 3. A modified DG start involving idling and gradual acceleration to synchronous speed may be used for this SR as recommended by the manufacturer. When modified start procedures are not used, the time, speed, voltage, and frequency tolerances of SR 3.8.1.7 must be met. ----- Verify each DG starts from standby conditions and achieves speed ≥ 900 rpm, steady state voltage ≥ 3785 V and ≤ 4400 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.3 · -----NOTES-----</p> <ol style="list-style-type: none"> 1. DG loadings may include gradual loading as recommended by the manufacturer. 2. Momentary transients outside the load range do not invalidate this test. 3. This Surveillance shall be conducted on only one DG at a time. 4. This SR shall be preceded by and immediately follow without shutdown a successful performance of SR 3.8.1.2 or SR 3.8.1.7. <p>-----</p> <p>Verify each DG is synchronized and loaded and operates for ≥ 60 minutes at a load ≥ 2370 kW and ≤ 2610 kW.</p>	<p>31 days</p>
<p>SR 3.8.1.4 Verify each day tank contains ≥ 250 gal of fuel oil.</p>	<p>31 days</p>
<p>SR 3.8.1.5 Check for and remove accumulated water from each day tank.</p>	<p>31 days</p>
<p>SR 3.8.1.6 Verify the fuel oil transfer system operates to transfer fuel oil from storage tanks to the day tank.</p>	<p>31 days</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.7 · -----NOTE----- All DG starts may be preceded by an engine prelube period. ----- Verify each DG starts from standby condition and achieves:</p> <ul style="list-style-type: none"> a. in \leq 10 seconds, speed \geq 900 rpm; and b. in \leq 13 seconds, voltage \geq 3785 V and \leq 4400 V, and frequency \geq 58.8 Hz and \leq 61.2 Hz. 	<p>184 days</p>
<p>SR 3.8.1.8 -----NOTE----- This Surveillance shall not be performed in MODE 1 or 2. ----- Verify automatic and manual transfer of AC power sources from the normal offsite circuit to the alternate required offsite circuit and manual transfer from the alternate offsite circuit to the delayed access circuit.</p>	<p>18 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.9 -----NOTES----- 1.This Surveillance shall not be performed in MODE 1 or 2. 2.If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. ----- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 63 Hz; b. Within 2.4 seconds following load rejection, the voltage is ≥ 3785 V and ≤ 4400 V; and c. Within 2.4 seconds following load rejection, the frequency is ≥ 58.8 Hz and ≤ 61.2 Hz.</p>	<p>18 months</p>
<p>SR 3.8.1.10 -----NOTE----- This Surveillance shall not be performed in MODE 1 or 2. ----- Verify each DG operating at a power factor ≤ 0.87 does not trip and voltage is maintained ≤ 6200 V during and following a load rejection of ≥ 2370 kW and ≤ 2610 kW.</p>	<p>18 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3:8.1.11 -----NOTES-----</p> <ol style="list-style-type: none"> 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. <p>-----</p> <p>Verify on an actual or simulated loss of offsite power signal:</p> <ol style="list-style-type: none"> a. De-energization of emergency buses; b. Load shedding from emergency buses; c. DG auto-starts from standby condition and: <ol style="list-style-type: none"> 1. energizes permanently connected loads in ≤ 10 seconds. 2. energizes auto-connected loads through auto-transfer sequencing timers. 3. maintains steady state voltage ≥ 3785 V and ≤ 4400 V. 4. maintains steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected loads for ≥ 5 minutes. 	<p>18 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.12 · -----NOTES-----</p> <ol style="list-style-type: none"> 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1 or 2. <p>-----</p> <p>Verify on an actual or simulated Safety Injection signal each DG auto-starts from standby condition and:</p> <ol style="list-style-type: none"> a. In \leq 13 seconds after auto-start and during tests, achieves voltage \geq 3785 V and \leq 4400 V; b. In \leq 13 seconds after auto-start and during tests, achieves frequency \geq 58.8 Hz and \leq 61.2 Hz; c. Operates for \geq 5 minutes; d. Permanently connected loads are energized from the alternate offsite power source; and e. Emergency loads are auto-connected through the ESF load sequencing timers to the alternate offsite power source. 	<p>18 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.13 -----NOTE----- This Surveillance shall not be performed in MODE 1 or 2. -----</p> <p>Verify each DG's automatic trips are bypassed when the diesel engine trip cutout switch is in the cutout position and the DG is aligned for automatic operation except:</p> <ul style="list-style-type: none"> a. Engine overspeed; b. Generator differential current; and c. Low lube oil pressure; 	<p>18 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.14 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not be performed in MODE 1 or 2. <p>-----</p> <p>Verify each DG operating at a power factor ≤ 0.87 operates for ≥ 24 hours:</p> <ol style="list-style-type: none"> a. For ≥ 2 hours loaded ≥ 2625 kW and ≤ 2890 kW; and b. For the remaining hours of the test loaded ≥ 2370 kW and ≤ 2610 kW. 	<p>18 months</p>
<p>SR 3.8.1.15 -----NOTES-----</p> <ol style="list-style-type: none"> 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 2 hours loaded ≥ 2370 kW and ≤ 2610 kW. <p>Momentary transients outside of load range do not invalidate this test.</p> <ol style="list-style-type: none"> 2. All DG starts may be preceded by an engine prelube period. <p>-----</p> <p>Verify each DG starts and achieves:</p> <ol style="list-style-type: none"> a. in ≤ 10 seconds, speed ≥ 900 rpm; and b. in ≤ 13 seconds, voltage ≥ 3785 V, and ≤ 4400 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz. 	<p>18 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.16 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -----</p> <p>Verify each DG:</p> <ul style="list-style-type: none"> a. Synchronizes with offsite power source while loaded with emergency loads upon a simulated restoration of offsite power; b. Transfers loads to offsite power source; and c. Returns to ready-to-load operation. 	<p>18 months</p>
<p>SR 3.8.1.17 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -----</p> <p>Verify, with a DG operating in test mode and connected to its bus, an actual or simulated Safety Injection signal overrides the test mode by:</p> <ul style="list-style-type: none"> a. Opening the auxiliary transformer breaker; and b. Automatically sequencing the emergency loads onto the DG. 	<p>18 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.18 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ----- Verify each ESF and auto-transfer load sequencing timer is within its limits.</p>	<p>18 months</p>
<p>SR 3.8.1.19 -----NOTES----- 1. All DG starts may be preceded by an engine prelube period. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ----- Verify on an actual or simulated loss of offsite power signal in conjunction with an actual or simulated Safety Injection signal: a. De-energization of emergency buses; b. Load shedding from emergency buses; and c. DG auto-starts from standby condition and: 1. energizes permanently connected loads in ≤ 10 seconds, 2. energizes auto-connected emergency loads through load sequencing timers, 3. achieves steady state voltage ≥ 3785 V and ≤ 4400 V, 4. achieves steady state frequency ≥ 58.8 Hz and ≤ 61.2 Hz, and 5. supplies permanently connected and auto-connected emergency loads for ≥ 5 minutes.</p>	<p>18 months</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.1.20 -----NOTE----- All DG starts may be preceded by an engine prelube period. -----</p> <p>Verify when started simultaneously from standby condition, each DG achieves:</p> <p>a. in \leq 10 seconds, speed \geq 900 rpm; and</p> <p>b. in \leq 13 seconds, voltage \geq 3785 V and \leq 4400 V, and frequency \geq 58.8 Hz and \leq 61.2 Hz.</p>	<p>10 years</p>

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

- a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown";
- b. One diesel generator (DG) capable of supplying the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10; and
- c. One supply train of the diesel fuel oil (DFO) transfer system.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION
A. One required offsite circuit inoperable.	-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required Class 1E AC electrical power distribution subsystem de-energized as a result of Condition A. -----	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	<u>OR</u> A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION
<p>A. (continued)</p>	<p>A.2.2 Suspend movement of irradiated fuel assemblies.</p> <p><u>AND</u></p> <p>A.2.3 Initiate action to suspend operations involving positive reactivity additions.</p> <p><u>AND</u></p> <p>A.2.4 Initiate action to restore required offsite power circuit to OPERABLE status.</p>	<p>Immediately</p> <p>Immediately</p> <p>Immediately</p>
<p>B. The required DG inoperable.</p> <p><u>OR</u></p> <p>The required supply train of the DFO transfer system inoperable.</p>	<p>B.1 Suspend CORE ALTERATIONS.</p> <p><u>AND</u></p> <p>B.2 Suspend movement of irradiated fuel assemblies.</p> <p><u>AND</u></p> <p>B.3 Initiate action to suspend operations involving positive reactivity additions.</p> <p><u>AND</u></p> <p>B.4 Initiate action to restore required DG to OPERABLE status.</p>	<p>Immediately</p> <p>Immediately</p> <p>Immediately</p> <p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.2.1 -----NOTE----- The following SRs are not required to be performed: SR 3.8.1.3, SR 3.8.1.9 through SR 3.8.1.11, SR 3.8.1.14 through SR 3.8.1.16, and SR 3.8.1.18 (for auto-transfer timers). ----- For AC sources required to be OPERABLE, the following SRs of Specification 3.8.1, "AC Sources - Operating," are applicable; SR 3.8.1.1 through SR 3.8.1.7, SR 3.8.1.9 through SR 3.8.1.11, SR 3.8.1.14 through SR 3.8.1.16, and SR 3.8.1.18 (for auto-transfer timers)</p>	<p>In accordance with applicable SRs</p>

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, Starting Air, and Turbocharger Air Assist

LCO 3.8.3 The stored diesel fuel oil, lube oil, starting air, and turbocharger air assist subsystems shall be within limits for each required diesel generator (DG). The fuel level for the stored diesel fuel oil shall be within the following limits:

- a. Combined storage of $\geq 65,000$ gal for two units in MODES 1, 2, 3, and 4; or
- b. Combined storage of
 - 1. $\geq 33,000$ gal for one unit (if any) in MODES 1, 2, 3, and 4; and
 - 2. $\geq 26,000$ gal for each unit in MODES 5 and 6.

-----NOTE-----
 The performance of diesel fuel oil tank cleaning requires one fuel oil storage tank to be removed from service to be drained and cleaned. During this time, the fuel oil storage requirement for one unit operation in MODES 1, 2, 3, and 4 and one unit operation in MODE 6 with at least 23 feet of water above the reactor vessel flange or with the reactor vessel defueled is $\geq 35,000$ gallons. The tank being cleaned may be inoperable for up to 10 days. For the duration of the tank cleaning, temporary onsite fuel oil storage of $\geq 24,000$ gallons will be maintained. Prior to removal of a tank from service, the offsite circuits required by LCO 3.8.1 or 3.8.2 will be verified to be OPERABLE.

APPLICABILITY: When associated DG(s) is required to be OPERABLE.

ACTIONS

-----NOTE-----
 Separate Condition entry is allowed for each DG or fuel oil storage tank, except for Condition A.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Combined fuel level in storage tanks not within limits.	A.1.1 Verify combined fuel oil level $\geq 29,000$ gallons for each unit operating in MODES 1,2,3,or 4.	Immediately
	<u>AND</u> A.1.2 Verify combined fuel oil level $\geq 23,000$ gallons for each unit operating in MODES 5 or 6.	Immediately
	<u>AND</u>	(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Restore fuel oil level to within limits.	48 hours
B. Both units in MODE 1, 2, 3, or 4 with lube oil inventory < 650 gal and > 610 gal. <u>OR</u> One unit in MODE 1, 2, 3, or 4 and one unit in MODE 5 or 6 with lube oil inventory < 590 gal and > 520 gal.	B.1 Restore lube oil inventory to within limits.	48 hours
C. One or more fuel oil storage tanks with stored fuel oil total particulates not within limit.	C.1 Restore fuel oil total particulates within limits.	7 days
D. One or more fuel oil storage tanks with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days
E. One or more DGs with both starting air receiver pressures < 180 psig and ≥ 150 psig.	E.1 Restore one starting air receiver pressure per DG to ≥ 180 psig.	48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>F. One or more DGs with turbocharger air assist air receiver pressure < 180 psig and ≥ 150 psig.</p>	<p>F.1 Restore turbocharger air assist air receiver pressure to ≥ 180 psig.</p>	<p>48 hours</p>
<p>G. Required Action and associated Completion Time not met.</p> <p><u>OR</u></p> <p>One or more DG's fuel oil, lube oil, turbocharger air assist, or starting air subsystem not within limits for reasons other than Condition A, B, C, D, E or F.</p>	<p>G.1 Declare associated DG inoperable.</p>	<p>Immediately</p>
<p>H. Required Action and associated Completion Time of Condition A, B, C, or D not met.</p> <p><u>OR</u></p> <p>Fuel oil storage tanks or lube oil not within limits for reasons other than Conditions A, B, C, or D.</p>	<p>H.1 Declare all DGs on associated unit(s) inoperable.</p> <p><u>AND, if associated unit is in MODES 1, 2, 3, or 4,</u></p> <p>H.2 Be in MODE 3.</p> <p><u>AND</u></p> <p>H.3 Be in MODE 5.</p>	<p>Immediately</p> <p>6 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.3.1 Verify fuel oil storage tanks contain combined storage within limits.	31 days
SR 3.8.3.2 Verify lubricating oil inventory is \geq 650 gal.	31 days
SR 3.8.3.3 Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4 Verify each DG has at least one air start receiver with a pressure is \geq 180 psig.	31 days
SR 3.8.3.5 Check for and remove accumulated water from each fuel oil storage tank.	31 days
SR 3.8.3.6 Verify each DG turbocharger air assist air receiver pressure is \geq 180 psig.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 Three Class 1E DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours
B. More than one full capacity charger receiving power simultaneously from a single 480 V vital bus. <u>OR</u> Any DC bus not receiving power from its associated AC electrical power distribution subsystem.	B.1 Restore the DC electrical power subsystem to a configuration wherein each charger is powered from its associated 480 volt vital bus.	14 days
C. Required Action and Associated Completion Time not met.	C .1 Be in MODE 3. <u>AND</u> C .2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.4.1 Verify battery terminal voltage is ≥ 130 V on float charge.	31 days
SR 3.8.4.2 Verify no visible corrosion at battery terminals and connectors. OR Verify battery connection resistance is $\leq 150 \times 10^{-6}$ ohm $\leq 150 \times 10^{-6}$ ohm for inter-cell connections, $\leq 150 \times 10^{-6}$ ohm for inter-rack connections, and $\leq 150 \times 10^{-6}$ ohm terminal connections.	92 days
SR 3.8.4.3 Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.	18 months
SR 3.8.4.4 Remove visible terminal corrosion, verify battery cell to cell and terminal connections are clean and tight, and are coated with anti-corrosion material.	18 months
SR 3.8.4.5 Verify battery connection resistance is $\leq 150 \times 10^{-6}$ ohm for inter-cell connections, $\leq 150 \times 10^{-6}$ ohm for inter-rack connections, and $\leq 150 \times 10^{-6}$ ohm for terminal connections.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.4.6 Verify each battery charger supplies ≥ 400 amps at ≥ 130 V for ≥ 4 hours.	18 months
SR 3.8.4.7 -----NOTES----- 1. The modified performance discharge test in SR 3.8.4.8 may be performed in lieu of the service test in SR 3.8.4.7 . 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. ----- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.8.4.8 -----NOTE----- This Surveillance shall not be performed in MODE 1, 2, 3, or 4. -----</p> <p>Verify battery capacity is \geq 80% of the manufacturer's rating when subjected to a performance discharge test or a modified performance discharge test.</p>	<p>60 months</p> <p><u>AND</u></p> <p>18 months when battery shows degradation or has reached 85% of expected life for the application.</p> <p><u>AND</u></p> <p>24 months when battery has reached 85% of the expected life with capacity \geq 100% of manufacturer's rating</p>

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 The Class 1E DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10. "Distribution Systems - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.5.1 -----NOTE----- The following SRs are not required to be performed: SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. ----- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.4 SR 3.8.4.7 SR 3.8.4.2 SR 3.8.4.5 SR 3.8.4.8. SR 3.8.4.3 SR 3.8.4.6</p>	<p>In accordance with applicable SRs</p>

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LC0 3.8.6 Battery cell parameters for the three Class 1E batteries shall be within the limits of Table 3.8.6-1.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more batteries with one or more battery cell parameters not within Category A or B limits.</p>	<p>A.1 Verify pilot cell electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.</p>	<p>1 hour</p>
	<p><u>AND</u></p> <p>A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits.</p>	<p>24 hours</p> <p><u>AND</u></p> <p>Once per 7 days thereafter</p>
	<p><u>AND</u></p> <p>A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.</p>	<p>31 days</p>

(continued)

ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>B. Required Action and associated Completion Time of Condition A not met.</p> <p><u>OR</u></p> <p>One or more batteries with average electrolyte temperature of the representative cells < 60°F.</p> <p><u>OR</u></p> <p>One or more batteries with one or more battery cell parameters not within Category C values.</p>	<p>B.1 Declare associated battery inoperable.</p>	<p>Immediately</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.8.6.1 Verify battery cell parameters meet Table 3.8.6-1 Category A limits.</p>	<p>31 days</p>

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.6.2 Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days <u>AND</u> Once within 7 days after a battery discharge < 118 V <u>AND</u> Once within 7 days after a battery overcharge > 145 V
SR 3.8.6.3 Verify average electrolyte temperature of representative cells is $\geq 60^{\circ}\text{F}$.	92 days

Table 3.8.6-1 (page 1 of 1)
Battery Cell Parameters Requirements

PARAMETER	CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL	CATEGORY B: LIMITS FOR EACH CONNECTED CELL	CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL
Electrolyte Level	> Minimum level indication mark, and $\leq \frac{1}{4}$ inch above maximum level indication mark ^(a)	> Minimum level indication mark, and $\leq \frac{1}{4}$ inch above maximum level indication mark ^(a)	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 V	≥ 2.13 V	> 2.07 V
Specific Gravity ^{(b)(c)}	≥ 1.195	≥ 1.190 <u>AND</u> Average of all connected cells > 1.200	Not more than 0.020 below average of all connected cells <u>AND</u> Average of all connected cells ≥ 1.190

- (a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.
- (b) Corrected for electrolyte temperature and level. Level correction is not required, however, when battery charging current is < 2 amps when on float charge.
- (c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters - Operating

LCO 3.8.7 Four Class 1E Vital 120 V UPS inverters shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 -----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating" with any vital 120 V AC bus de-energized. ----- Restore inverter to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.7.1 Verify correct inverter voltage and alignment to required AC vital buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 The Class 1E UPS Inverters shall be OPERABLE to support the onsite Class 1E 120 VAC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required inverters inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>	(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Initiate action to restore required inverters to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.8.1 Verify correct inverter voltage, and alignments, to required AC vital buses.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 The required Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One AC electrical power distribution subsystems inoperable.	A.1 Restore AC electrical power distribution subsystem to OPERABLE status.	8 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
B. One 120 VAC vital bus subsystem inoperable.	B.1 Restore 120 VAC vital bus subsystem to OPERABLE status.	2 hours <u>AND</u> 16 hours from discovery of failure to meet LCO
C. One DC electrical power distribution subsystem inoperable.	C.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours <u>AND</u> 16 hours from discovery of failure to meet LCO (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u> D.2 Be in MODE 5.	36 hours
E. Two required Class 1E AC, DC, or 120 VAC vital buses with inoperable distribution subsystems that result in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.9.1 Verify correct breaker alignments and voltage to required AC, DC, and 120 VAC vital bus electrical power distribution subsystems.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 - The necessary portion of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6.
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required. AC, DC, or 120 VAC vital bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
<u>AND</u>		
A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately	
<u>AND</u>	(continued)	

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

-----NOTE-----
While this LCO is not met, entry into MODE 6 from MODE 5 is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS. <u>AND</u>	Immediately
	A.2 Suspend positive reactivity additions. <u>AND</u>	Immediately
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.1.1 Verify boron concentration is within the limit specified in COLR.	72 hours

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1 Suspend CORE ALTERATIONS except for latching control rod drive shafts and friction testing of individual control rods..	Immediately
	<u>AND</u> A.2 Suspend positive reactivity additions.	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1 Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	<u>AND</u> B.2 Perform SR 3.9.1.1.	Once per 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.3.1 Perform CHANNEL CHECK.	12 hours
SR 3.9.3.2 -----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	18 months

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

- LCO 3.9.4 The containment penetrations shall be in the following status:
- a. The equipment hatch closed and held in place by four bolts;
 - b. One door in each air lock closed; and
 - c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:
 - 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or
 - 2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation valve.

-----NOTE-----
 Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS,
 During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u> A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.4.1 Verify each required containment penetration is in the required status except for containment penetrations that are open under administrative controls.	7 days
SR 3.9.4.2 Verify each required containment purge and exhaust ventilation isolation valves actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

LC0 3.9.5 One RHR loop shall be OPERABLE and in operation.

-----NOTE-----
The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

The required RHR loop maybe removed from operation for ≤ 2 hours per 8 hour period for performance of leak testing the RHR suction isolation valves provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RHR loop requirements not met.	A.1 Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	<u>AND</u>	
	A.2 Suspend loading irradiated fuel assemblies in the core.	Immediately
	<u>AND</u>	
	A.3 Initiate action to satisfy RHR loop requirements.	Immediately
	<u>AND</u>	
	A.4 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.9.5.1 With the reactor subcritical less than 57 hours, verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm.</p> <p><u>OR</u></p> <p>With the reactor subcritical for 57 hours or more, verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 1300 gpm.</p>	<p>12 hours</p>

3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

-----NOTE-----
While this LCO is not met, entry into a MODE or other specified condition in the APPLICABILITY is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore required RHR loops to OPERABLE status.	Immediately
	<u>OR</u> A.2 Initiate action to establish \geq 23 ft of water above the top of reactor vessel flange.	Immediately
B. No RHR loop in operation.	B.1 Suspend operations involving a reduction in reactor coolant boron concentration. <u>AND</u>	Immediately (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Initiate action to restore one RHR loop to operation. <u>AND</u> B.3 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	Immediately 4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.6.1 With the reactor subcritical less than 57 hours, verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm. <u>OR</u> With the reactor subcritical for 57 hours or more, verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 1300 gpm.	12 hours
SR 3.9.6.2 Verify correct breaker alignment and indicated power available to the required RHR pump that is not in operation.	7 days

3.9 REFUELING OPERATIONS

3.9.7 Refueling Cavity Water Level

LCO 3.9.7 Refueling cavity water level shall be maintained \geq 23 ft above the top of reactor vessel flange.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refueling cavity water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.7.1 Verify refueling cavity water level is \geq 23 ft above the top of reactor vessel flange.	24 hours

4.0 DESIGN FEATURES

4.1 Site Location

The DCPD site consists of approximately 750 acres which are adjacent to the Pacific Ocean in San Luis Obispo County, California, and is approximately twelve (12) miles west-southwest of the city of San Luis Obispo.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core locations.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control rod material shall be silver, indium, and cadmium, as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

(continued)

4.0 DESIGN FEATURES

- b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2.3 of the FSAR;
- c. A nominal 11 inch center to center distance between fuel assemblies placed in the fuel storage racks

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

- a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;
- b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1.1 of the FSAR;
- c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1.1.1 of the FSAR; and
- d. A nominal 22 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pools are designed and shall be maintained to prevent inadvertent draining of the pool below elevation 133 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1324 fuel assemblies.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

- 5.1.1 The Vice President, Diablo Canyon Operations and Plant Manager, hereafter called Plant Manager, shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The Plant Manager or his designee, hereafter called Plant Manager, shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

- 5.1.2 The Shift Foreman (SFM) shall be responsible for the control room command function. During any absence of the SFM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SFM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.
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5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

- a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR Update;
- b. The Plant Manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;
- c. The Senior Vice President and General Manager - Nuclear Power Generation shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and
- d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

5.2.2 Unit Staff

The unit staff organization shall include the following:

- a. A non-licensed operator shall be assigned to each reactor containing fuel with a total of three non-licensed operators required for both units.

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

- b. Shift crew composition may be one less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a and 5.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.
- c. A health physics technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.
- d. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed Senior Reactor Operators (SROs), licensed Reactor Operators (ROs), health physic technicians, nuclear operators, and key maintenance personnel).

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40 hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time;
2. An individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48 hour period, nor more than 72 hours in any 7 day period, all excluding shift turnover time;
3. A break of at least 8 hours should be allowed between work periods, including shift turnover time;

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized in advance by the Plant Manager or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Superintendent Plant Manager or his designee to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

- e. The Operations Director shall hold an SRO license.
 - f. The Shift Technical Advisor (STA) shall provide advisory technical support to the SFM in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA position shall be manned in MODES 1,2,3, and 4 unless an individual with a SRO license meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.
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5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 for comparable positions, except for the Radiation Protection Director who shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 2, April 1987 for Radiation Protection Manager. The licensed ROs and SROs shall also meet or exceed the minimum qualifications of 10 CFR Part 55 and the supplemental requirements specified in Section A of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

A retraining and replacement training program for the plant staff shall be maintained under the direction of a designated member of the facility staff and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

- 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:
- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
 - b. The emergency operating procedures required to implement the applicable requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33 and responses to the subject NUREGs;
 - c. Quality assurance for effluent and environmental monitoring;
 - d. Fire Protection Program implementation; and
 - e. All programs specified in Specification 5.5.
-

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and
- b. The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification 5.6.2 and Specification 5.6.3.

Licensee initiated changes to the ODCM:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
 1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and
 2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
- b. Shall become effective after the approval of the Plant Manager; and
- c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

(continued)

5.5 Programs and Manuals

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include portions of Recirculation Spray, Safety Injection, Chemical and Volume Control, Residual Heat Removal, RCS Sample, and Liquid and Gaseous Radwaste Treatment Systems. The program shall include the following:

- a. Preventive maintenance and periodic visual inspection requirements; and
- b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

- a. Training of personnel;
- b. Procedures for sampling and analysis; and
- c. Provisions for maintenance of sampling and analysis equipment.

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;
- b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2, to 10 CFR 20.1001-20.2402;
- c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;
- d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;
- e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with methodology and parameters in the ODCM. Determination of projected dose contributions for radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;
- f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

- g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the following:
 - 1. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and
 - 2. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ.
- h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;
- i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and
- j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section 5.2 and 5.3, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Not Used

(continued)

5.5 Programs and Manuals (continued)

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), conduct a qualified in-place UT examination over the volume from the inner-bore of the flywheel to the circle of one-half the outer radius or conduct a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels once every ten years coinciding with the Inservice Inspection schedule as required by ASME Section XI.

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

- a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

<u>ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities</u>	<u>Required Frequencies for performing inservice testing activities</u>
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

- b. The provisions of SR 3.0.2 are applicable to the above required frequencies for performing inservice testing activities;

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program (continued)

- c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and
- d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Tube Surveillance Program

SG tube integrity shall be demonstrated by performance of the following augmented inservice inspection program.

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test frequencies.

- a. SG Sample Selection and Inspection - SG tube integrity shall be determined during shutdown by selecting and inspecting at least the minimum number of SGs specified in Table 5.5.9-1.
- b. SG Tube Sample Selection and Inspection - The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of SG tubes shall be performed at the frequencies specified in Specification 5.5.9.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.d. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all SGs; the tubes selected for these inspections shall be selected on a random basis except:
 - 1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;
 - 2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each SG shall include:

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

- a) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),
 - b) Tubes in those areas where experience has indicated potential problems, and
 - c) A tube inspection (pursuant to Specification 5.5.9.d.1.h) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.
3. The tubes selected as the second and third samples (if required by Table 5.5.9-2) during each inservice inspection may be subjected to a partial tube inspection provided:
- a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and
 - b) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

<u>Category</u>	<u>Inspection Results</u>
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

c. Inspection Frequencies - The above required inservice inspections of SG tubes shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;
2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.5.9-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.c.1. The interval may then be extended to a maximum of once per 40 months; and
3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:
 - a) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13; or
 - b) A seismic occurrence greater than the Double Design Earthquake, or
 - c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or
 - d) A main steam line or feedwater line break.

d. Acceptance Criteria

1. As used in this Specification:

(continued)

5.5 Programs and Manuals (continued)

- a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;
- b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;
- c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;
- d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;
- e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;
- f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;
- g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of a Double Design Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.c.3.c, above;

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

- h) Tube Inspection means an inspection of the SG tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg; and
- i) Preservice Inspection means an inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial Power Operation using the equipment and techniques expected to be used during subsequent inservice inspections.

- 2. The SG tube integrity shall be determined after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.9-2.

e. Reports

The contents and frequency of reports concerning the SG tube surveillance program shall be in accordance with Specification 5.6.10.

(continued)

5.5 Programs and Manuals (continued)

TABLE 5.5.9-1

MINIMUM NUMBER OF STEAM GENERATORS (SGs) TO BE
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
	Two	Three	Four	Two	Three	Four
No. of SGs per Unit						
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One ¹			One ¹	One ²	One ³

TABLE NOTATIONS

1. The inservice inspection may be limited to one SG on a rotating schedule encompassing 3 N % of the tubes (where N is the number of SGs in the plant) if the results of the first or previous inspections indicate that all SGs are performing in a like manner. Note that under some circumstances, the operating conditions in one or more SGs may be found to be more severe than those in other SGs. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.
2. The other SG not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.
3. Each of the other two SGs not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

(continued)

5.5 Programs and Manuals (continued)

TABLE 5.5.9-2
STEAM GENERATOR (SG) TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
			C-3	Perform action for C-3 result of first sample	C-2	Plug defective tubes
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to 50.72(b)(2) of 10 CFR Part 50	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	C-3	Perform action for C-3 result of first sample
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to 50.72(b)(2) of 10 CFR Part 50	N/A	N/A

$$S=3 \frac{N}{n}$$

Where N is the number of SGs in the unit, and n is the number of SGs inspected during an inspection

(continued)

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified below and in accordance with Regulatory Guide 1.52, Revision 2, ASME N510-1 1980, and ASTM D3803-1989.

- a. Demonstrate for each of the ESF systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance with ASME N510-1980 at the system flowrate specified below $\pm 10\%$ at least once per operating cycle.

ESF Ventilation System	Flowrate
Control Room	2100 cfm
Auxiliary Building	73,500 cfm
Fuel Handling Building	35,750 cfm

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- b. Demonstrate for each of the ESF systems that an in-place test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with ASME N510-1980 at the system flowrate specified below $\pm 10\%$ at least once per operating cycle.

(continued)

5.5 Programs and Manuals (continued)

ESF Ventilation System	Flowrate
Control Room	2100 cfm
Auxiliary Building	73,500 cfm
Fuel Handling Building	35,750 cfm

- c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and at the relative humidity specified below. Laboratory testing shall be completed at least once per 18 months and after every 720 hours of charcoal operation.

ESF Ventilation System	Penetration	RH
Control Room	1.0 %	70%
Auxiliary Building	6.0 %	70%
Fuel Handling Building	4.3 %	95%

- d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested in accordance with ASME N510- 1980 at the system flowrate specified below $\pm 10\%$ at least once per operating cycle.

ESF Ventilation System	Delta P	Flowrate
Control Room	3.5 in. WG	2100 cfm
Auxiliary Building	3.7 in. WG	73,500 cfm
Fuel Handling Building	4.1 in. WG	35,750 cfm

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- e. Demonstrate that the charcoal pre-heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ASME N510-1980 at least once per operating cycle.

ESF Ventilation System	Wattage
Control Room	5 ± 1 kW
Auxiliary Building	50 ± 5 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in temporary unprotected outdoor liquid storage tanks.

The gaseous radioactivity quantities shall be determined following the methodology in Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure." The liquid radwaste quantities shall be maintained such that 10 CFR Part 20 limits are met.

The program shall include:

- a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);
- b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and
- c. A surveillance program to ensure that the quantity of radioactivity contained in temporary outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals (continued)

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
 - 1. an API gravity or an absolute specific gravity within limits,
 - 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
 - 3. a clear and bright appearance with proper color; or water and sediment content within limits.
- b. Other properties for ASTM 2D fuel oil are analyzed within 31 days following sampling and addition to storage tanks; and
- c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276, Method A .
- d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
 - 1. a change in the TS incorporated in the license; or

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program (continued)

2. a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

- a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
- b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;
- c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
- d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (SFDP) (continued)

- a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
- b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995."

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 47 psig.

The maximum allowable containment leakage rate, L_a , at P_a , shall be 0.10 % of containment air weight per day.

Leakage rate acceptance criteria are:

- a. Containment overall leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60 L_a$ for the Type B and Type C tests and $\leq 0.75 L_a$ for Type A tests;
- b. Air lock testing acceptance criteria are:
 - 1) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program (continued)

- 2) For each door, leakage rate is $\leq 0.01 L$, when pressurized to ≥ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

- 1) Training of personnel, and
- 2) Procedures for monitoring.

5.5.18 Containment Polar and Turbine Building Cranes

A program which will ensure that: 1) the position of the containment polar cranes precludes jet impingement from a postulated pipe rupture; and 2) the operation of the turbine building cranes is consistent with the restrictions associated with the current Hosgri seismic analysis of the turbine building. This program shall include the following:

- 1) Training of personnel, and
 - 2) Procedures for the containment polar and turbine building cranes operation.
-
-

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrem and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

-----NOTE-----
A single submittal may be made for a multiple unit station. The submittal shall combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

(continued)

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:
1. Shutdown Bank Insertion Limits for Specification 3.1.5,
 2. Control Bank Insertion Limits for Specification 3.1.6,
 3. Axial Flux Difference for Specification 3.2.3,
 4. Heat Flux Hot-Channel Factor, $K(Z)$ and $W(Z) - F_q(z)$ (F_q^{RTP} Specification 3.2.1),
 5. RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}^N$ ($F_{\Delta H}^{RTP}$ and $PF_{\Delta H}$ for Specification 3.2.2),
 6. SHUTDOWN MARGIN values in Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8,
 7. Moderator Temperature Coefficient limits in Specification 3.1.3, and
 8. Refueling Boron Concentration limits in Specification 3.9.1.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
1. WCAP-10216-P-A, Revision 1A, Relaxation of Constant Axial Offset Control F, Surveillance Technical Specification, February 1994 (Westinghouse Proprietary).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (Westinghouse Proprietary).
 3. WCAP-8385, Power Distribution Control and Load Following Procedures, September 1974 (Westinghouse Proprietary).
 4. WCAP-10054-P-A, Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code, August 1985. (Westinghouse Proprietary), and
 5. WCAP-10266-P-A, Revision 2 with Addenda, The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, December 14, 1987. (Westinghouse Proprietary).
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, Low Temperature Overpressure Protection (LTOP) arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and
 2. Specification 3.4.12, "Low Temperature Overpressure Protection System."

(continued)

5.6 Reporting Requirements (continued)

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

- b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
1. The analytical methods used to determine the RCS pressure and temperature limits were developed in accordance with:
 - 10 CFR 50, Appendix G and H
 - Regulatory Guide 1.99, Revision 2
 - NUREG-0800, Standard Review Plan Section 5.3.2
 - Branch Technical Position MTEB 5-2
 - ASME B&P Code Section III, Appendix G
 - ASME B&P Code, Section XI, Appendix A
 - WCAP-14040-NP-A, Section 2.2
 2. LTOP limits (Power Operated Relief Valves (PORV) pressure relief setpoint and LTOP enable temperature) were developed in accordance with:
 - NUREG-0800, Standard Review Plan Section 5.2.2
 - Branch Technical Position RSB 5-2
 - 10 CFR 50, Appendix G and H
 - Regulatory Guide 1.99, Revision 2
 - Branch Technical Position MTEB 5-2
 - WCAP-14040-NP-A, Section 2.2
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not Used

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not Used

5.6 Reporting Requirements (continued)

5.6.10 Steam Generator (SG) Tube Inspection Report

- a. Within 15 days following the completion of each inservice inspection of SG tubes, the number of tubes plugged in each SG shall be reported to the Commission.
 - b. The complete results of the SG tube inservice inspection shall be submitted to the Commission in a report within 12 months following completion of the inspection. This Special Report shall include:
 - 1) Number and extent of tubes inspected,
 - 2) Location and percent of wall-thickness penetration for each indication of an imperfection, and
 - 3) Identification of tubes plugged.
 - c. Results of SG tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.
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5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr at 30 cm, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

- a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.
- b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.
- c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the radiation protection supervision in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr at 30 cm shall be provided with locked or continuously guarded doors to prevent inadvertent entry and the keys shall be maintained under the administrative control of the Shift Foreman on duty or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP that shall specify the dose rate levels in the immediate work areas and the maximum allowable stay times for individuals in those areas. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

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5.7 High Radiation Area

- 5.7.3 For individual high radiation areas with radiation levels of > 1000 mrem/hr at 30 cm, accessible to personnel, that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that cannot be continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.
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JLS Conversion to Improved Technical Specifications Diablo Canyon Power Plant

Docket # 50-275
Accession # 9706230042
Date 6/2/97 of Ltr
Regulatory Docket File

Improved TS Bases



B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and steam generator safety valves prevents violation of the reactor core SLs.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the following fuel design criteria:

- a. There must be at least 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and
- b. The hot fuel pellet in the core must not experience centerline fuel melting.

The Reactor Trip System setpoints (Ref. 2), in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

- a. High pressurizer pressure trip;
- b. Low pressurizer pressure trip;
- c. Overtemperature ΔT trip;
- d. Overpower ΔT trip;
- e. Power Range Neutron Flux trip; and
- f. Steam generator safety valves.

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref.5) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The curves provided in Figure B 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the

(continued)

BASES

SAFETY LIMITS
(continued)

enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The curves are based on enthalpy hot channel factor limits provided in the COLR.

The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT
VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 7.2.
3. WCAP-8746-A, March 1977.
4. WCAP-9273-NP-A, July 1985.
5. FSAR, Chapter 15.

(continued)

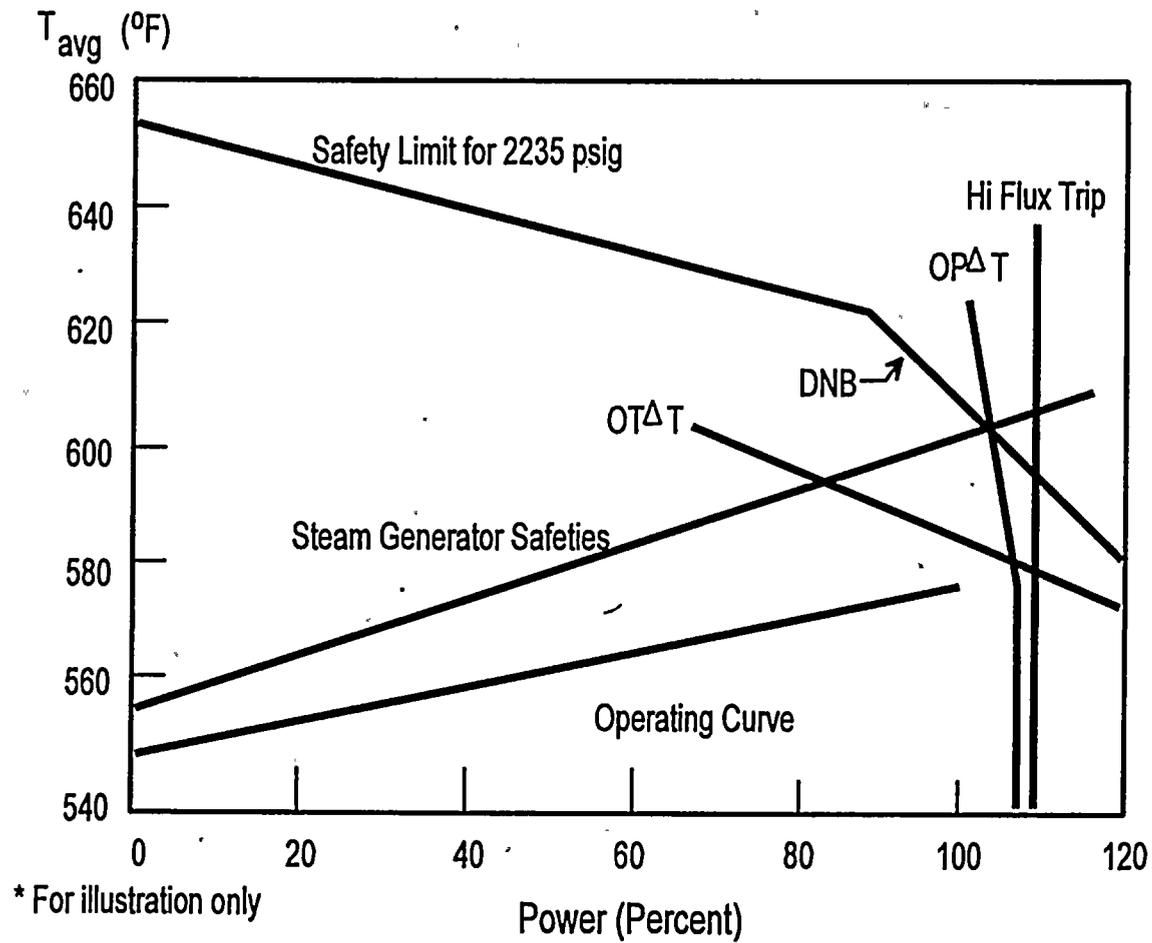


Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components were hydrostatically tested at 150% (3750 psia) (Ref. 5) of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and A00s. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

- a. Pressurizer power operated relief valves (PORVs);
- b. Steam line relief valve;
- c. Steam Dump System;
- d. Reactor Control System;
- e. Pressurizer Level Control System; or
- f. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under [USAS, Section B31.1 (Ref. 6)] is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, or the reactor vessel is sufficiently vented, making it unlikely that the RCS can be pressurized.

SAFETY LIMIT
VIOLATIONS

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

(continued)

BASES

SAFETY LIMIT
VIOLATIONS
(continued)

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
 2. Not used
 3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
 4. 10 CFR 100.
 5. Westinghouse report SD-117, "Structural Analysis of Reactor Coolant Loop/Support System for Diablo Canyon Nuclear Generating Station, Unit No. 1, February, 1975.
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B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs LCO 3.0.1 through LCO 3.0.6 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1 LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2 LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

- a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and
- b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not restricted by the Completion Time. In this case, compliance with

(continued)

BASES

LCO 3.0.2
(continued)

the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time other conditions exist which result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable, and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

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BASES

- a. An associated Required Action and Completion Time is not met and no other Condition applies; or
- b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit.

Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

- a. The LCO is now met.
- b. A Condition exists for which the Required Actions have now been performed.

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BASES

LCO 3.0.3
(continued)

- c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of Specification 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.15, "Fuel Storage Pool Water Level." LCO 3.7.15 has an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.15 are not met while in MODE 1, 2, 3, or 4, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.15 of "Suspend movement of irradiated fuel assemblies in the fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

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BASES

LCO 3.0.4 LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

- a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and
- b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply

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BASES

LCO 3.0.4
(continued)

MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. In some cases (e.g., where a plant-specific review has concluded that specific restriction on MODE changes should be included) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s) to allow the performance of required testing to demonstrate:

- a. The OPERABILITY of the equipment being returned to service; or
- b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

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BASES

LCO 3.0.5
(continued)

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing .

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

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BASES

LCO 3.0.6
(continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCO 3.1.8, allows specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the

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BASES

LCO 3.0.7
(continued)

MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS LCO 3.0.7 requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1 SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

- a. The systems or components are known to be inoperable, although still meeting the SRs; or
- b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply.

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BASES

SR 3.0.1
(continued)

Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is those surveillances performed in accordance with the Containment Leakage Rate Testing Program.

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BASES

SR 3.0.2
(continued)

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing

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BASES

SR 3.0.3
(continued)

the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES

(continued)

BASES

or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit.

SR 3.0.4
(continued)

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or component to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions

(continued)

BASES

needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

SR 3.0.4
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SR 3.0.4 is only applicable when entering MODE 4 from MODE 5, MODE 3 from MODE 4, Mode 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4.

The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1), the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all shutdown and control rods, assuming that the single rod cluster assembly of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable control assemblies and soluble boric acid in the Reactor Coolant System (RCS). The Rod Control System can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, the Rod Control System, together with the boration system, provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn. The Chemical and Volume Control System can control the soluble boron concentration to compensate for fuel depletion during operation and all xenon burnout reactivity changes and can maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured, assuming that core reactivity is within design limit of LCO 3.1.2, by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCO 3.1.6, "Control Bank Insertion Limits." When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

APPLICABLE
SAFETY ANALYSIS

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analysis establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth rod stuck out on scram. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

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BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are not exceeded. This is done by ensuring that:

- a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;
- b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 200 cal/gm average fuel pellet enthalpy at the hot spot in irradiated fuel for the rod ejection accident, Ref. 5); and
- c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accidents for the SDM requirements are the main steam line break (MSLB) and inadvertent boron dilution accidents, as described in the FSAR (Refs. 2 and 3). In addition to the limiting MSLB transient, the SDM requirement is also used in the analyses of the following events:

- a. Inadvertent boron dilution;
- b. An uncontrolled rod withdrawal from subcritical or low power condition; and
- c. Start of an inactive reactor coolant pump (RCP); and
- d. Rod ejection.

The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected steam generator (SG), and consequently the RCS. This results in a reduction of the reactor coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. As RCS temperature decreases, the severity of an MSLB decreases until the MODE 5 value is reached. The most limiting MSLB, with respect to potential fuel damage before a reactor trip occurs, is a guillotine break of a main steam line inside containment initiated at the end of core life with RCS T_{avg} equal to 547°F. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating RCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration.

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BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

The startup of an inactive RCP in MODES 1 or 2 is precluded. In MODE 3, the startup of an inactive RCP can not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent start is less than half the minimum required SDM. Startup of an idle RCP cannot, therefore, produce a return to power from the hot standby condition.

SDM satisfies Criterion 2 of the 10CFR50.36(c)(2)(ii). Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration. LCO The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be sufficient. The required SDM is specified in the COLR.

APPLICABILITY

In MODE 2 with $k_{eff} < 1.0$ and in MODES 3, 4, and 5 the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits."

BASES

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the borated water source should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the refueling water storage tank. The operator should borate with the best source available for the plant conditions.

SURVEILLANCE
REQUIREMENTSSR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODES 2 (with $k_{eff} < 1.0$), 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects (SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth):

- a. RCS boron concentration;
- b. Control and shutdown rod position;
- c. RCS average temperature;
- d. Fuel burnup based on gross thermal energy generation;
- e. Xenon concentration;
- f. Samarium concentration; and
- g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.1.1 (continued)

performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
 2. FSAR, Chapter 15, Section 15.4.2.1.
 3. FSAR, Chapter 15, Section 15.2.4.
 4. 10 CFR 100.
 5. FSAR, Chapter 15, Section 15.4.6.1.6.
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BASES

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Core Reactivity

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as burnable absorbers producing zero net reactivity. Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations and that the calculational models used to generate the safety analysis are adequate.

In order to achieve the required fuel cycle energy output, the uranium enrichment, in the new fuel loading and in the fuel remaining from the previous cycle, provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical at RTP and moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, whatever neutron poisons (mainly xenon and samarium) are present in the fuel, and the RCS boron concentration.

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BASES

BACKGROUND
(continued)

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the RCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE
SAFETY ANALYSES

The acceptance criteria for core reactivity are that the reactivity balance limit ensures plant operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the RCS boron concentration requirements for reactivity control during fuel depletion as well as providing inputs to the safety analysis.

The comparison between measured and predicted initial core reactivity provides a validation of the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value when deemed necessary shall be performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually

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BASES

APPLICABLE
SAFETY ANALYSES
(Continued)

monitored and evaluated as core conditions change during the cycle.

Core reactivity satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily altered once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the reactivity balance of $\pm 1\% \Delta k/k$ has been established based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

When measured core reactivity is within $1\% \Delta k/k$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis. Core reactivity and control rod worth measurements are required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, control rod shuffling).

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BASES

ACTIONS

A.1 and A.2

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve and the boron concentration requirement for SDM may be renormalized and power operation may continue. If operational restrictions or additional SRs are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 72 hours is adequate for preparing whatever operating restrictions or surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then the boration required by LCO 3.1.1 Required Action A.1 would occur. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made, considering that other core conditions are fixed or stable, including control rod position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note. The Note indicates that the normalization (adjustment, only if necessary) of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (QPTR, AFD, etc.) for prompt indication of an anomaly.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 28, and GDC 29.
 2. FSAR, Chapter 15.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Reload cores are designed so that the beginning of cycle (BOC) MTC is less than zero when THERMAL POWER is at RTP. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional fixed distributed poisons to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure that the MTC does not exceed the EOC limit.

The limitations on MTC are provided to ensure that the value of this coefficient remains within the limiting conditions assumed in the FSAR accident and transient analyses.

If the LCO limits are not met, the unit response during transients may not be as predicted. The core could violate criteria that prohibit a return to criticality, or the departure from nucleate boiling ratio criteria of the approved correlation may be violated, which could lead to a loss of the fuel cladding integrity.

The SRs for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits, since this coefficient changes slowly, due principally to the reduction in RCS boron concentration associated with fuel burnup.

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BASES

APPLICABLE
SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

- a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and
- b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

The FSAR, Chapter 15 (Ref. 2), contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding (Ref. 3).

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from either zero (Ref. 2) or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches a boron concentration equivalent to 300 ppm at an equilibrium, all rods out, RTP condition. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

The most negative MTC value, equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR accident analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) adding margin to this value to account for the largest difference in MTC observed between an EOC, all rods withdrawn, RATED THERMAL POWER condition and an envelope of those most adverse conditions of moderator temperature and pressure, rods

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

inserted to their insertion limits, axial power skewing, and xenon concentration that can occur in normal operation within Technical Specification limits and lead to a significantly more negative EOC MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR accident analyses into the limiting EOC MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by adding an allowance for burnup and soluble boron concentration changes to the limiting EOC MTC value.

MTC satisfies Criterion 2 of the 10CFR50.36(c)(2)(ii). Even though it is not directly observed and controlled from the control room, MTC is considered an initial condition process variable because of its dependence on boron concentration.

LCO

LCO 3.1.3 requires the MTC to be within specified limits of the COLR to ensure that the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation.

Assumptions made in safety analyses require that the MTC be less positive than a given upper bound and more positive than a given lower bound. The MTC is most positive near BOC when core reactivity and required boron concentration are at their maximum values; this upper bound must not be exceeded. This maximum upper limit is evaluated near BOC, all rods out (ARO), hot zero power conditions. At EOC the MTC takes on its most negative value, when the lower bound becomes important. This LCO exists to ensure that both the upper and lower bounds are not exceeded.

During operation, therefore, the conditions of the LCO can only be ensured through measurement. The Surveillance checks at BOC and EOC on MTC provide confirmation that the MTC is behaving as anticipated so that the acceptance criteria are met.

The LCO establishes a maximum positive value that cannot be exceeded. The BOC positive limit and the EOC negative limit are established in the COLR to allow specifying limits for each particular cycle. This permits the unit to take advantage of improved fuel management and changes in unit operating schedule.

APPLICABILITY

Technical Specifications place both LCO and SR values on MTC, based on the safety analysis assumptions described above.

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2 with the reactor critical, the upper limit must also be maintained to ensure that startup and subcritical accidents (such as the uncontrolled control rod assembly or group withdrawal)

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BASES

APPLICABILITY
(continued)

will not violate the assumptions of the accident analysis. The lower MTC limit must be maintained in MODES 2 and 3, in addition to MODE 1, to ensure that cooldown accidents will not violate the assumptions of the accident analysis. In MODES 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents using the MTC as an analysis assumption are initiated from these MODES.

ACTIONS

A.1

If the upper MTC limit is violated, administrative withdrawal limits for control banks must be established to maintain the MTC within its limits. The MTC becomes more negative with control bank insertion and decreased boron concentration. A Completion Time of 24 hours provides enough time for evaluating the MTC measurement and computing the required bank withdrawal limits.

As cycle burnup is increased, the RCS boron concentration will be reduced. The reduced boron concentration causes the MTC to become more negative. Using physics calculations, the time-in-cycle life at which the calculated MTC will meet the LCO requirement can be determined. At this point in core life Condition A no longer exists. The unit is no longer in the Required Action, so the administrative withdrawal limits are no longer in effect.

B.1

If the required administrative withdrawal limits are not established within 24 hours, the unit must be brought to MODE 2 with $k_{eff} < 1.0$ to prevent operation with an MTC that is more positive than that assumed in safety analyses.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

C.1

Exceeding the EOC MTC limit means that the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the EOC MTC limit is exceeded, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 4 within 12 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

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BASES

SURVEILLANCE
REQUIREMENTSSR 3.1.3.1

This SR requires measurement of the MTC at BOC prior to entering MODE 1 in order to demonstrate compliance with the most positive MTC LCO. Meeting the BOC limit prior to entering MODE 1 ensures that the limit will also be met at higher power levels.

The BOC MTC value for ARO will be inferred from isothermal temperature coefficient measurements obtained during the physics tests after refueling. The ARO value can be directly compared to the BOC MTC limit of the LCO. If required, measurement results and predicted design values can be used to establish administrative withdrawal limits for control banks.

SR 3.1.3.2

In similar fashion, the LCO demands that the MTC be less negative than the specified value for EOC full power conditions. This measurement may be performed at any THERMAL POWER, but its results must be extrapolated to the conditions of RTP and all banks withdrawn in order to make a proper comparison with the LCO value. Because the RTP MTC value will gradually become more negative with further core depletion and boron concentration reduction, a 300 ppm SR value of MTC should necessarily be less negative than the EOC LCO limit. The 300 ppm SR value is sufficiently less negative than the EOC LCO limit value to ensure that the LCO limit will be met when the 300 ppm Surveillance criterion is met.

SR 3.1.3.2 is modified by three Notes that include the following requirements:

1. The SR is required to be performed once each cycle within 7 effective full power days (EFPDs) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm.
2. If the 300 ppm Surveillance limit is exceeded, it is possible that the EOC limit on MTC could be reached before the planned EOC. Because the MTC changes slowly with core depletion, the Frequency of 14 effective full power days is sufficient to avoid exceeding the EOC limit.
3. The Surveillance limit for RTP boron concentration of 60 ppm is conservative. If the measured MTC at 60 ppm is less negative than the 60 ppm Surveillance limit, the EOC limit will not be exceeded because of the gradual manner in which MTC changes with core burnup.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 11.
2. FSAR, Chapter 15.

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BASES

REFERENCES
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3. WCAP-9273-A , "Westinghouse Reload Safety Evaluation Methodology." July 1985.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM.

Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately $\frac{5}{8}$ inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously and are moved in a staggered fashion, but always within one step of each other. All control banks contain two rod groups. Two shutdown banks (A and B) contain two rod groups and the remaining two shutdown banks (C and D) contain one rod group.

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control

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BASES

BACKGROUND
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bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position (continued) indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or $\pm 5/8$ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half accuracy. The DRPI system is capable of monitoring rod position within at least ± 12 steps with either full accuracy or half accuracy.

APPLICABLE
SAFETY ANALYSIS

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing rod inoperability or misalignment are that:

- a. There be no violations of:
 1. specified acceptable fuel design limits, or
 2. Reactor Coolant System (RCS) pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control or shutdown rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck

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APPLICABLE
ANALYSIS
(continued)

fully withdrawn. This condition requires an evaluation to SAFETY determine that sufficient reactivity worth is held in the rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment (Ref. 3). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from bank D inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 4).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned rod is allowed if the heat flux hot channel factor ($F_0(Z)$) and the nuclear enthalpy hot channel factor ($F_{\Delta H}^N$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and $F_0(Z)$ and $F_{\Delta H}^N$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_0(Z)$ and $F_{\Delta H}^N$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements (i.e., trippability to meet SDM) are separate from the alignment requirements, which ensure that the RCCAs and banks maintain the correct power distribution and rod alignment. The rod OPERABILITY requirement is satisfied provided the rod will fully insert in the required time assumed in the

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LCO
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safety analyses. Rod control malfunctions that result in the inability to move a rod (e.g., rod urgent failures), but do not impact trippability, do not necessarily result in rod inoperability.

The requirement to maintain the rod alignment to within plus or minus 12 steps of their group step counter demand position is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

The requirement to maintain rod alignment is met by comparing individual rod DRPI indication and bank demand position indication to be within plus or minus 12 steps. If one of these position indicators become inoperable, the conditions of this LCO are still met by compliance with LCO 3.1.7.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are typically fully inserted and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 2 with $k_{eff} < 1.0$, 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

ACTIONS

A.1.1 and A.1.2

When one or more rods are inoperable (i.e., untrippable), there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration and restoring SDM.

With an inoperable rod(s), This ACTION provides for verification of SDM, this is most simply accomplished by verifying rod insertion limits are met. Additionally, actions could include calculation of the current SDM and boration to meet limits specified in the COLR or proceed to MODE 3. These actions are

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consistent with those specified in LCO 3.1.5 and LCO 3.1.6.

A rod is considered trippable if it was demonstrated OPERABLE during the last performance of SR 3.1.4.2 and met the rod drop time criteria during the last performance of SR 3.1.4.3.

In this situation, SDM verification must account for the absence of the negative reactivity of the untrippable rod(s), as well as the rod of maximum worth.

A.2

If the inoperable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable (i.e., OPERABLE). If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group demand position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be fully inserted and control bank C must be inserted to approximately 100 steps.

Power operation may continue with one RCCA OPERABLE (i.e. trippable) but misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary,

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BASES

ACTIONS
(continued)

aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F_{\Delta H}^N$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 3). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q(Z)$ and $F_{\Delta H}^N$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q(Z)$ and $F_{\Delta H}^N$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

The following accident analyses require re-evaluation for continued operation with a misaligned rod:

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary Coolant System Pipe Rupture

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ACTIONS
(continued)

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection).

C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

D.1.1 and D.1.2

More than one rod becoming misaligned from its group demand position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

Additionally, the requirements of LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," apply if the misaligned rods are not within the required insertion limits.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows

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BASES

the operator to detect a rod that is beginning to deviate from its expected position. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each rod would result in radial or axial power tilts, or oscillations. Exercising each individual rod every 92 days provides confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between or during required performances of SR 3.1.4.2 (determination of rod OPERABILITY by movement), if a rod(s) is discovered to be immovable, but remains trippable, the rod(s) is considered to be OPERABLE. At any time, if a rod(s) is immovable, a determination of the trippability (OPERABILITY) of the rod(s) must be made, and appropriate action taken.

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature $\geq 500^{\circ}\text{F}$ to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. FSAR, Chapter 15, Section 15.2.3.
4. FSAR, Chapter 15, Section 4.2.3.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available ejected rod worth, SDM and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs consists of two groups that are moved in a staggered fashion, but always within one step of each other. All four control banks contain two rod groups. Two shutdown banks (A and B) contain two rod groups and the remaining two shutdown banks (C and D) consist of a single group. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks

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BACKGROUND
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must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE
SAFETY ANALYSIS

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN - (SDM)") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion safety limits and inoperability or misalignment is that:

- a. There be no violations of:
 1. specified acceptable fuel design limits, or
 2. RCS pressure boundary integrity; and
- b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 3).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

The shutdown bank insertion limits are defined in the COLR.

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BASES

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. The applicability in MODE 2 begins at initial control bank withdrawal, during an approach to criticality, and continues throughout MODE 2, until all control bank rods are again fully inserted by reactor trip or by shutdown. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks are, typically, fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODES 2 with $k_{eff} < 1.0$, 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

ACTIONS

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time. Additionally, the requirements of LCO 3.1.4, "Rod Group Alignment Limits," apply if one or more shutdown rods are not within the required alignment limits.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.
 2. 10 CFR 50.46.
 3. FSAR, Chapter 15, Section 15.4.3.2.4.
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(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Control Bank Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously and are moved in a staggered fashion, but always within one step of each other. Two shutdown banks (C and D) consist of a single group. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control bank insertion limits are specified in the COLR. The control banks are required to be at or above the insertion limit lines.

The COLR also indicates how the control banks are moved in an overlap pattern. Overlap is the distance traveled together by two control banks. The control banks are used for precise reactivity control of the reactor. The positions of the control banks can be controlled manually, or automatically by the Rod Control System. They are capable of altering reactivity very quickly (compared to borating or diluting).

The power density at any point in the core must be limited, so that the fuel design criteria are maintained. Together, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," LCO 3.1.6, "Control Bank Insertion Limits," LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," provide limits on control component operation and on monitored process variables, which ensure that the core operates within the fuel design criteria.

(continued)

BASES

BACKGROUND
(continued)

The shutdown and control bank insertion and alignment limits, AFD, and QPTR are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the control bank insertion limits control the reactivity that could be added in the event of a rod ejection accident, and the shutdown and control bank insertion limits ensure the required SDM is maintained assuming LCO 3.1.2, "Core Reactivity" is met for core reactivity.

Operation within the subject LCO limits will prevent fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a loss of coolant accident (LOCA), loss of flow, ejected rod, or other accident requiring termination by a Reactor Trip System (RTS) trip function.

APPLICABLE
SAFETY ANALYSES

The shutdown and control bank insertion limits, AFD, and QPTR LCOs are required to prevent power distributions that could result in fuel cladding failures in the event of a LOCA, loss of flow, ejected rod, or other accident requiring termination by an RTS trip function.

The acceptance criteria for addressing shutdown and control bank insertion limits and inoperability or misalignment are that:

- a. There be no violations of:
 1. specified acceptable fuel design limits, or
 2. Reactor Coolant System pressure boundary integrity;
and
- b. The core remains subcritical after accident transients.

As such, the shutdown and control bank insertion limits affect safety analysis involving core reactivity and power distributions (Ref. 3).

The SDM requirement is ensured by limiting the control and shutdown bank insertion limits so that allowable inserted worth of the RCCAs is such that sufficient reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes the maximum worth RCCA remains fully withdrawn upon trip (Ref. 4).

Operation at the insertion limits or AFD limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPTR present. Operation at the insertion limit may also indicate the maximum ejected RCCA worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected RCCA worths.

The control and shutdown bank insertion limits ensure that safety analyses assumptions for SDM, ejected rod worth, and power distribution peaking factors are preserved (Ref. 5).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The insertion limits satisfy Criterion 2 of 10CFR50.36(c)(2)(ii), in that they are initial conditions assumed in the safety analysis.

LCO

The limits on control banks sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion is available on trip. The overlap between control banks provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during control bank motion.

APPLICABILITY

The control bank sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2 with $k_{eff} \geq 1.0$. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES.

The applicability requirements have been modified by a Note indicating the LCO requirements are suspended during the performance of SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the control bank to move below the LCO limits, which would violate the LCO.

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2

When the control banks are outside the acceptable insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

- a. Reducing power to be consistent with rod position; or
- b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required within 1 hour, since the SDM in MODES 1 and 2 normally ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1, "SHUTDOWN MARGIN - (SDM)") has been upset. If control banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in the Bases for SR 3.1.1.1.

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits. Failure of sequence or overlap support

(continued)

BASES

ACTIONS
(continued)

equipment does not require entering the ACTIONS as long as sequence and overlap limits are maintained.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlap limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

Additionally, the requirements of LCO 3.1.4, "Rod Group Alignment Limits," apply if one or more control rods are not within the required alignment limits.

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 3, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

Verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. The verification of compliance with the sequence and overlap limits specified in the COLR consists of an observation that the static rod positions of those control banks not fully withdrawn from the core are within the limits specified in the COLR. Bank sequence and overlap must also be maintained during rod movement, implicit within the LCO. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.
 2. 10.CFR 50.46.
 3. FSAR, Chapter 4, Section 4.3.2.4.
 4. FSAR, Chapter 4, Section 4.3.2.4.
 5. WCAP-9273-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 .
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B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the rod position indicators to determine rod positions and thereby ensure compliance with the rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

Mechanical or electrical failures may cause a rod to become inoperable or to become misaligned from its group. Rod inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown.

Limits on rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their rod drive mechanisms. The RCCAs are divided among four control banks and four shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or $\pm \frac{5}{8}$ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

(continued)

BASES

BACKGROUND
(continued)

The DRPI System provides a highly accurate indication of actual rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data system fails, the DRPI will go on half accuracy. The DRPI system is capable of monitoring rod position within at least ± 12 steps with either full accuracy or half accuracy.

APPLICABLE
SAFETY ANALYSIS

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the bank sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The rod position indicator channels satisfy Criterion 2 of 10CFR50.36(c)(2)(ii). The rod position indicators monitor rod position, which is an initial condition of the accident.

LCO

LCO 3.1.7 specifies that the DRPI System and the Bank Demand Position Indication System be OPERABLE for each rod. For the rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

- a. The DRPI System on either full accuracy or half accuracy indicates within 12 steps of the group step counter demand position as required by LCO 3.1.4, "Rod Group Alignment Limits"; and
- b. The Bank Demand Indication System has been reset in the fully inserted position, fully withdrawn position or to the DRPI System.

The 12 step agreement limit between the Bank Demand Position Indication System and the DRPI System indicates that the Bank Demand Position Indication System is adequately calibrated, and can be used for indication of the measurement of rod bank position.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single rod, ensures high confidence that the position uncertainty of the corresponding rod

(continued)

BASES

LCO
(continued)

group is within the assumed values used in the analysis (that specified rod group insertion limits).

These requirements ensure that rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator and each demand position indicator per bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

When one DRPI per group fails, the position of the rod may still be determined indirectly by use of the movable incore detectors. The Required Action may also be ensuring at least once per hours that f_0 satisfies LCO 3.2.1, F_{RH} satisfies LCO 3.2.2, and SDM is within the limits provided in the COLR, provided the nonindicating rods have not been moved. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of C.1 or C.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

A.2

Reduction of THERMAL POWER to $\leq 50\%$ RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3).

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to $\leq 50\%$ RTP from full

(continued)

BASES

ACTIONS
(continued)

power conditions without challenging plant systems and allowing for rod position determination by Required Action A.1 above.

B.1 and B.2

When more than one DRPI per group fail, additional actions are necessary to ensure that acceptable power distribution limits are maintained, minimum SDM is maintained, and the potential effects of rod misalignment on associated accident analyses are limited. The indirect position determination available via movable incore detectors will minimize the potential for rod misalignment.

The position of the rods can be determined indirectly by use of the movable incore detectors. The Required Action may also be satisfied by ensuring at least once per 8 hours that F_0 satisfies LCO 3.2.1, F_H satisfies LCO 3.2.2, and SHUTDOWN MARGIN is within the limits provided is the COLR, provided that the nonindicating rods have not moved. Verification of RCCA position once per 8 hours is adequate for allowing continued full power operation for a limited, 24 hour period, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small. The 24 hour Allowed Outage Time provides sufficient time to troubleshoot and restore the DRPI system to operation while avoiding the plant challenges associated with a shutdown without full rod position indication.

Based on operating experience, normal power operation does not require excessive rod movement. If one or more control rods has been significantly moved, the Required Action of C.1 or C.2 below is required.

C.1 and C.2

These Required Actions clarify that when one or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1 and A.2 or B.1 are still appropriate but must be initiated promptly under Required Action C.1 to begin indirectly verifying that these rods are still properly positioned, relative to their group positions.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to $\leq 50\%$ RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at $> 50\%$ RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions using the movable incore detectors.

D.1.1 and D.1.2

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal

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BASES

ACTIONS
(continued)

power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apart within the allowed Completion Time of once every 8 hours is adequate.

D.2

Reduction of THERMAL POWER to $\leq 50\%$ RTP puts the core into a condition where rod position is not significantly affecting core peaking factors (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions D.1.1 and D.1.2 or reduce power to $\leq 50\%$ RTP.

E.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.1.8.1

Verification that the DRPI agrees with the demand position within 12 steps ensures that the DRPI is operating correctly. Verification at 24, 48, 120, and 228 steps withdrawn for the control and shutdown banks provides assurance that the DRPI is operating correctly over the full range of indication.

This surveillance is performed prior to reactor criticality after each removal of the reactor head, since there is potential for unnecessary plant transients if the SR were performed with the reactor at power.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13.
 2. FSAR, Chapter 15.
 3. WCAP-10216-P-A, Rev. 1A, "Relaxation of Constant Axial Offset Control and F_0 Surveillance Technical Specification," February 1994.
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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to (Ref. 3):

- a. Ensure that the facility has been adequately designed;
- b. Validate the analytical models used in the design and analysis;
- c. Verify the assumptions used to predict unit response;
- d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and
- e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality, during startup, during low power operations, during power ascension, at high power, and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions and that the core can be operated as designed.

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS required for reload fuel cycles in MODE 2 typically include:

(continued)

BASES

BACKGROUND
(continued)

- a. Critical Boron Concentration.
- b. Control Rod Worthand.
- c. Isothermal Temperature Coefficient (ITC).

These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

APPLICABLE
SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

Reload fuel cycle PHYSICS TESTS are performed in accordance with Technical Specification requirements, fuel vendor guidelines, and established industry practices. Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.4, "Moderator Temperature Coefficient (MTC)," LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to $\leq 5\%$ RTP, the reactor coolant temperature is kept $\geq 531^\circ\text{F}$, and SDM is within the limits provided in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of 10 CFR 50.36(c)(2)(ii).

Reference 6 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion

(continued)

BASES

LCO
(continued)

limits. Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met. The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

- a. RCS lowest operating loop average temperature is ≥ 531 °F;
 - b. SDM is within the limits provided in the COLR; and
 - c. THERMAL POWER is $\leq 5\%$ RTP.
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APPLICABILITY

This LCO is applicable when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is $> 5\%$ RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest operating loop's T_{avg} is < 531 °F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with an operating loop's temperature below 531°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is

(continued)

BASES

ACTIONS (continued) reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The required power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each OPERABLE power range and intermediate range channels within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest operating loop T_{avg} is $\geq 531^{\circ}\text{F}$ will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

Verification that the THERMAL POWER is $\leq 5\%$ RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 1 hour during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

Verification that the SDM is within limits specified in the COLR ensures that, for the specific RCCA and RCS temperature manipulations performed during PHYSICS TESTS, the plant is not operating in a condition that could invalidate the safety analysis assumptions.

The SDM for physics testing during tests where traditional SDM monitoring techniques are not adequate, is determined for the most restrictive test based on design calculations. Plant conditions are monitored during these tests to verify adequate SDM.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

(continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
 2. 10 CFR 50.59.
 3. Regulatory Guide 1.68, Revision 2, August, 1978.
 4. Not used.
 5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.
 6. WCAP-11618, including Addendum 1, April 1989.
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B 3.2 POWER DISTRIBUTION LIMITS

 B 3.2.1 Heat Flux Hot Channel Factor ($F_0(Z)$)

 BASES

BACKGROUND

The purpose of the limits on the values of $F_0(Z)$ is to limit the local (i.e., pellet) peak power density. The value of $F_0(Z)$ varies along the axial height (Z) of the core.

$F_0(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_0(Z)$ is a measure of the peak fuel pellet power within the reactor core.

During power operation, the global power distribution is limited by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT TILT POWER RATIO (QPTR)," which are directly and continuously measured process variables. These LCOs, along with LCO 3.1.6, "Control Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

$F_0(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution.

$F_0(Z)$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. The results of the power distribution map are analyzed to derive a measured value for $F_0(Z)$. These measurements are generally taken with the core at or near equilibrium conditions.

However, because this value represents an equilibrium condition, it does not include the variations in the value of $F_0(Z)$ that are present during nonequilibrium situations, such as load following.

To account for these possible variations, a transient $F_0(Z)$ is also calculated based on the steady state value of $F_0(Z)$. In this case, the steady state $F_0(Z)$ is adjusted by an elevation dependent factor, $W(Z)$, that accounts for the calculated transient conditions.

Core monitoring and control under nonsteady state conditions are accomplished by operating the core within the limits of the appropriate LCOs, including the limits on AFD, QPTR, and control rod insertion.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

This LCO's principal effect is to preclude core power distributions that could lead to violation of the following fuel design criterion:

During a large break loss of coolant accident (LOCA), the peak cladding temperature must not exceed 2200°F (Ref. 1).

Limits on $F_0(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, the LOCA peak cladding temperature is typically most limiting.

$F_0(Z)$ limits assumed in the LOCA analysis are typically limiting relative to (i.e., lower than) the $F_0(Z)$ limit assumed in safety analyses for other postulated accidents. Therefore, this LCO provides conservative limits for other postulated accidents.

$F_0(Z)$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

LCO

The Heat Flux Hot Channel Factor, $F_0(Z)$, shall be limited by the following relationships:

$$F_0(Z) \leq \frac{F_0^{RTP}}{P} K(Z) \quad \text{for } P > 0.5$$

$$F_0(Z) \leq \frac{F_0^{RTP}}{0.5} K(Z) \quad \text{for } P \leq 0.5$$

where: F_0^{RTP} is the $F_0(Z)$ limit at rated thermal power (RTP) provided in the COLR,

$K(Z)$ is the $F_0(Z)$ normalization factor for core height provided in the COLR, and

$$P = \frac{\text{THERMAL POWER}}{\text{RTP}}$$

The actual values of F_0^{RTP} and $K(Z)$ are given in the COLR.

For Relaxed Axial Offset Control operation, $F_0(Z)$ is approximated by $F_0^C(Z)$ and $F_0^H(Z)$. Thus, both $F_0^C(Z)$ and $F_0^H(Z)$ must meet the preceding limits on $F_0(Z)$.

An $F_0^C(Z)$ evaluation requires obtaining an incore flux map in MODE 1. From the incore flux map results we obtain the measured value ($F_0^M(Z)$) of $F_0(Z)$. The computed heat flux hot channel factor, $F_0^C(Z)$ is obtained by the equation:

$$F_0^C(Z) = F_0^M(Z) (1.03) (1.05)$$

where 1.03 is a factor that accounts for fuel manufacturing tolerances and 1.05 is a factor that accounts for flux map measurement uncertainty.

The expression for $F_0^H(Z)$ is:

$$F_0^H(Z) = F_0^C(Z) W(Z)$$

where $W(Z)$ is a cycle dependent function that accounts for power distribution transients encountered during normal operation. $W(Z)$ is included in the COLR.

The $F_0(Z)$ limits define limiting values for core power peaking that preclude peak cladding temperatures above 2200°F during either a large or small break LOCA.

(continued)

BASES

LCO

(continued)

The $F_0(Z)$ limits define limiting values for core power peaking that preclude peak cladding temperatures above 2200°F during either a large or small break LOCA.

This LCO requires operation within the bounds assumed in the safety analyses. If $F_0(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_0(Z)$ may produce unacceptable consequences if a design basis event occurs while $F_0(Z)$ is outside its specified limits.

If the power distribution measurements are performed at a power level less than 100% RTP, then the $F_0^C(Z)$ and $F_0^M(Z)$ values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with F_0^{RTP} assures compliance with the LCO at all power levels.

APPLICABILITY

The $F_0(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

Reducing THERMAL POWER by $\geq 1\%$ RTP for each 1% by which $F_0^C(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_0^C(Z)$ is $F_0^M(Z)$ multiplied by factors which account for manufacturing tolerances and measurement uncertainties. $F_0^M(Z)$ is the measured value of $F_0(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)

BASES

ACTIONS
(continued)A.2

A reduction of the Power Range Neutron Flux-High trip setpoints by $\geq 1\%$ for each 1% by which $F_0^c(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.3

Reduction in the Overpower ΔT trip setpoints by $\geq 1\%$ for each 1% by which $F_0^c(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.4

Verification that $F_0^c(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions. Inherent in this action is identification of the cause of the out of limit condition, and the correction of the cause, to the extent necessary to allow safe operation at the higher power level. The allowable power level is determined by extrapolating $F_0^c(Z)$. SR 3.2.1.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints.

B.1

If it is found that the maximum calculated value of $F_0(Z)$ that can occur during normal maneuvers, $F_0^H(Z)$, exceeds its specified limits, there exists a potential for $F_0^c(Z)$ to become excessively high if a normal operational transient occurs. Reducing the AFD by $\geq 1\%$ for each 1% by which $F_0^H(Z)$ exceeds its limit within the allowed Completion Time of 4 hours, restricts the axial flux distribution such that even if a transient occurred, core peaking factor limits are not exceeded.

ACTIONS

C.1

(continued)

BASES

(continued)

If Required Actions A.1 through A.4 or B.1 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving Mode 1). The note allows for power ascensions if the surveillances are not current. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_0^C(Z)$ and $F_0^W(Z)$ are within their specified limits after a power rise of more than 20% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because $F_0^C(Z)$ and $F_0^W(Z)$ could not have previously been measured for a reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_0^C(Z)$ and $F_0^W(Z)$ are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_0^C(Z)$ and $F_0^W(Z)$ following a power increase of more than 20%, ensures that they are verified within 24 hours from when equilibrium conditions are achieved at RTP (or any other level for extended operation). Equilibrium conditions are achieved when the core is sufficiently stable such that the uncertainties associated with the measurement are valid. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_0^C(Z)$ and $F_0^W(Z)$. The Frequency condition is not intended to require verification of these parameters after every 20% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 20% higher than that power at which F_0^C was last measured.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.2.1.1

Verification that $F_0^c(Z)$ is within its specified limits involves increasing $F_0^m(Z)$ to allow for manufacturing tolerance and measurement uncertainties in order to obtain $F_0^c(Z)$. Specifically, $F_0^m(Z)$ is the measured value of $F_0^c(Z)$ obtained from incore flux map results and $F_0^c(Z) = F_0^m(Z) (1.03) (1.05)$ (Ref. 2). $F_0^c(Z)$ is then compared to its specified limits.

The limit with which $F_0^c(Z)$ is compared varies inversely with power above 50% RTP and directly with a function called $K(Z)$ provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP (and meeting the 100% RTP $F_0(Z)$ limit) provides assurance that the $F_0^c(Z)$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by $\geq 20\%$ RTP since the last determination of $F_0^c(Z)$, another evaluation of this factor is required 24 hours after achieving equilibrium conditions at this higher power level to ensure that $F_0^c(Z)$ values are being reduced sufficiently with power increase to stay within the LCO limits.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 3.2.1.2

Because flux maps are taken in equilibrium conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z , is called $W(Z)$. Multiplying the measured total peaking factor, $F_0^c(Z)$, by $W(Z)$ gives the maximum $F_0(Z)$ calculated to occur in normal operation, $F_0^m(Z)$.

The limit with which $F_0^m(Z)$ is compared varies inversely with power and directly with the function $K(Z)$ provided in the COLR.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.2 (continued)

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_0^c(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

- a. Lower core region, from 0 to 15% inclusive; and
- b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. When $F_0^m(Z)$ is determined, an evaluation of the expression below is required to account for any increase to $F_0^c(Z)$ that may occur and cause the $F_0(Z)$ limit to be exceeded before the next required $F_0(Z)$ evaluation.

If the two most recent $F_0(Z)$ evaluations show an increase in the expression

$$\text{maximum over } z \quad \left[\frac{F_0^c(Z)}{K(Z)} \right]$$

it is required to meet the $F_0(Z)$ limit with the last $F_0^m(Z)$ increased by a factor ≥ 2 percent which is specified in the COLR, or to evaluate $F_0(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_0(Z)$ from exceeding its limit for any significant period of time without detection.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP or at a reduced power at any other time, and meeting the 100% RTP $F_0(Z)$ limit, provides assurance that the $F_0(Z)$ limit will be met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.1.2 (continued)

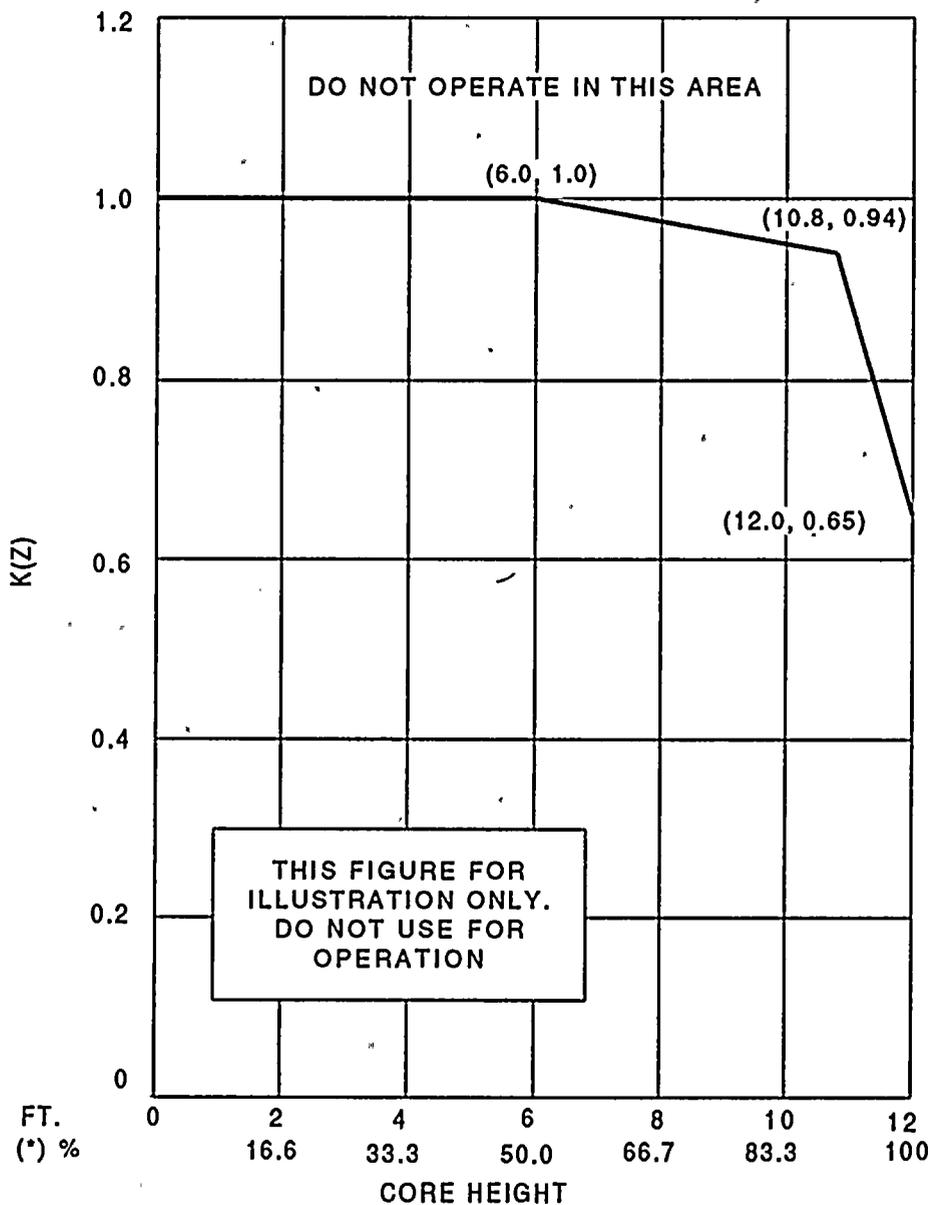
$F_0(Z)$ is verified at power levels \geq 20% RTP above the THERMAL POWER of its last verification, 24 hours after achieving equilibrium conditions to ensure that $F_0(Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is normally adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_0(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46, 1974.
 2. WCAP-7308-L-P-A, "Evaluation of Nuclear Hot Channel Factor Uncertainties," June 1988.
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*For core height of 12 feet

Figure B 3.2.1B-1 (page 1 of 1)
K(Z) - Normalized F₀(Z) as a Function of Core Height

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor (F_{AH}^N)BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during normal operation, operational transients, and any transient condition arising from events of moderate frequency analyzed in the safety analyses.

F_{AH}^N is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, F_{AH}^N is a measure of the maximum total power produced in a fuel rod. F_{AH}^N is sensitive to fuel loading patterns, bank insertion, and fuel burnup.

F_{AH}^N is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the power distribution map are analyzed to determine F_{AH}^N . This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables. Compliance with these LCOs, along with the LCOs governing shutdown and control rod insertion and alignment, maintains the core limits on power distribution on a continuous basis.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. All DNB limited transient events are assumed to begin with an F_{AH}^N value that satisfies the LCO requirements.

(continued)

BASES

BACKGROUND
(continued)

Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE
SAFETY ANALYSES

Limits on $F_{\text{DNB}}^{\text{N}}$ preclude core power distributions that exceed the following fuel design limits:

- a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition;
- b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;
- c. During an ejected rod accident, the average fuel pellet enthalpy at the hot spot in irradiated fuel must not exceed 280 cal/gm [Ref. 1]; and
- d. Fuel design limits required by GDC 26 (Ref. 2) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

The limits on $F_{\text{DNB}}^{\text{N}}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion applicable to a specific DNBR correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB condition.

The allowable $F_{\text{DNB}}^{\text{N}}$ limit increases with decreasing power level. This relationship between power and $F_{\text{DNB}}^{\text{N}}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs) of SL 2.1.1. Therefore, any DNB events in which the calculation of the core limits is modeled implicitly use

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

this variable value of $F_{\Delta H}^N$ in the analyses. Likewise, all transients that may be DNB limited are assumed to begin with a limiting initial $F_{\Delta H}^N$ as a function of power level defined by the $F_{\Delta H}^N$ limit equation in the COLR.

The LOCA safety analysis also uses $F_{\Delta H}^N$ as an input parameter. The Nuclear Heat Flux Hot Channel Factor ($F_0(Z)$) and the axial peaking factors are inserted directly into the LOCA safety analyses that verify the acceptability of the resulting peak cladding temperature [Ref. 3].

The fuel is protected in part by compliance with Technical Specifications, which ensure that the initial conditions assumed in the safety and accident analyses remain valid. The following LCOs ensure this: LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," LCO 3.1. 6, "Control Bank Insertion Limits," LCO 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)," and LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_0(Z)$)."

$F_{\Delta H}^N$ and $F_0(Z)$ are measured periodically using the movable incore detector system. Measurements are generally taken with the core at, or near, steady state conditions. Core monitoring and control under transient conditions (Condition 1 events) are accomplished by operating the core within the limits of the LCOs on AFD, QPTR, and Bank Insertion Limits.

$F_{\Delta H}^N$ satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

$F_{\Delta H}^N$ shall be maintained within the limits of the relationship provided in the COLR.

The $F_{\Delta H}^N$ limit is representative of the coolant flow channel with the maximum enthalpy rise. This channel has the highest probability for a DNB condition.

The limiting value of $F_{\Delta H}^N$, described by the equation contained in the COLR, is the design radial peaking factor used in the unit safety analyses.

A power multiplication factor in this equation includes an additional allowance for higher radial peaking factors from reduced thermal feedback and greater control rod insertion at low power levels. The limiting value of $F_{\Delta H}^N$ is allowed to increase by a cycle-dependent factor, $PF_{\Delta H}$, specified in the COLR for reduction in THERMAL POWER.

(continued)

BASES

LCO

(continued)

If the power distribution measurements are performed at a power level less than 100% RTP, then the F^N_{ΔH} values that would result from measurements if the core was at 100% RTP should be inferred from the available information. A comparison of these inferred values with F^{RIP}_{ΔH} assures compliance with the LCO at all power levels.

APPLICABILITY

The F^N_{ΔH} limits must be maintained in MODE 1 to preclude core power distributions from exceeding the fuel design limits for DNBR and PCT. Applicability in other modes is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the coolant to require a limit on the distribution of core power.

ACTIONS

A.1.1

With F^N_{ΔH} exceeding its limit, the unit is allowed 4 hours to restore F^N_{ΔH} to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring F^N_{ΔH} within its power dependent limit. When the F^N_{ΔH} limit is exceeded, the DNBR limit is not likely to be violated in steady state operation, because events that could significantly perturb the F^N_{ΔH} value (e.g., static control rod misalignment) are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs. Thus, the allowed Completion Time of 4 hours provides an acceptable time to restore F^N_{ΔH} to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time. The restoration of the peaking factor to within its limits by power reduction or control rod movement does not restore compliance with the LCO. Thus, this condition can not be exited until a valid surveillance demonstrates compliance with the LCO.

Condition A is modified by a Note that requires that Required Actions A.2 and A.3 must be completed whenever Condition A is entered. Thus, if power is not reduced because this Required Action is completed within the 4 hour time period, Required Action A.2 nevertheless requires another measurement and calculation of F^N_{ΔH} within 24 hours in accordance with SR 3.2.2.1.

(continued)

BASES

ACTIONS

A.1.1 (continued)

However, if power is reduced below 50% RTP, Required Action A.3 requires that another determination of F_{AH}^{N} must be done prior to exceeding 50% RTP, prior to exceeding 75%^{AH} RTP, and within 24 hours after reaching or exceeding 95% RTP; however, THERMAL POWER does not have to be reduced to comply with these requirements. In addition, Required Action A.2 is performed if power ascension is delayed past 24 hours.

A.1.2.1 and A.1.2.2

If the value of F_{AH}^{N} is not restored to within its specified limit either by adjusting a misaligned rod or by reducing THERMAL POWER, the alternative option is to reduce THERMAL POWER to < 50% RTP in accordance with Required Action A.1.2.1 and reduce the Power Range Neutron Flux-High to \leq 55% RTP in accordance with Required Action A.1.2.2. Reducing THERMAL POWER to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive.

The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints; however, for extended operations at the reduced power level, the reduced trip setpoints are required to protect against involving positive reactivity excursions. This is a sensitive operation that may inadvertently actuate the Reactor Protection System.

A.2

Once actions have been taken to restore F_{AH}^{N} to within its limits per Required Action A.1.1, or the power level has been reduced to < 50% RTP per Required Action A.1.2.1, an incore flux map (SR 3.2.2.1) must be obtained and the measured value of F_{AH}^{N} verified not to exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this

(continued)

BASES

ACTIONS

A.2 (continued)

task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate F^N_{AH}.

A.3

Verification that F^N_{AH} is within its specified limits after an out of limit occurrence ensures that the cause that led to exceeding the F^N_{AH} limit is identified, to the extent necessary, and corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the F^N_{AH} limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is \geq 95% RTP. SR 3.2.2.1 must be satisfied prior to increasing power above the extrapolated allowable power level or restoration of any reduced Reactor Trip System setpoints. When F^N_{AH} is measured at reduced power levels, the allowable power level is determined by evaluating F^N_{AH} for higher power levels.

This Required Action is modified by a Note that states that THERMAL POWER does not have to be reduced prior to performing this Action.

B.1

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, the plant must be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.2.2.1

SR 3.2.2.1 is modified by a Note. The Note applies during power ascensions following a plant shutdown (leaving Mode 1). The Note allows for power ascensions if the surveillances are not current.

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.2.2.1 (continued)

It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. Equilibrium conditions are achieved when the core is sufficiently stable such that uncertainties associated with the measurement are valid.

The value of $F_{\Delta H}^N$ is determined by using the movable incore detector system to obtain a flux distribution map. A data reduction computer program then calculates the maximum value of $F_{\Delta H}^N$ from the measured flux distributions. The measured value of $F_{\Delta H}^N$ must be multiplied by 1.04 to account for measurement uncertainty before making comparisons to the $F_{\Delta H}^N$ limit.

After each refueling, $F_{\Delta H}^N$ must be determined in MODE 1 prior to exceeding 75% RTP. This requirement ensures that $F_{\Delta H}^N$ limits are met at the beginning of each fuel cycle. Performing this Surveillance in MODE 1 prior to exceeding 75% RTP, or at a reduced power level at any other time, and meeting the 100% RTP $F_{\Delta H}^N$ limit, provides assurance that the $F_{\Delta H}^N$ limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the $F_{\Delta H}^N$ limit cannot be exceeded for any significant period of operation.

REFERENCES

1. Regulatory Guide 1.77, Rev. 0, May 1974.
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the amount of axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

Relaxed Axial Offset Control (RAOC) is a calculational procedure that defines the allowed operational space of the AFD versus THERMAL POWER. The AFD limits are selected by considering a range of axial xenon distributions that may occur as a result of large variations of the AFD. Subsequently, power peaking factors and power distributions are examined to ensure that the loss of coolant accident (LOCA), loss of flow accident, and anticipated transient limits are met. Violation of the AFD limits invalidate the conclusions of the accident and transient analyses with regard to fuel cladding integrity.

The AFD is monitored on an automatic basis using the plant process computer, which has an AFD monitor alarm. The computer determines the 1 minute average of each of the OPERABLE excore detector outputs and provides an alarm message if the AFD for two or more OPERABLE excore channels is outside its specified limits.

Although the RAOC defines limits that must be met to satisfy safety analyses, typically an operating scheme, is used to control axial power distribution in day to day operation (Ref. 1). This requires that the AFD be controlled within a narrow tolerance band around a burnup dependent target to minimize the variation of axial peaking factors and axial xenon distribution during unit maneuvers.

This operating space is typically smaller and lies within the RAOC operating space. Control within this operating space constrains the variation of axial xenon distributions and axial power distributions. RAOC calculations assume a wide range of xenon distributions and then confirm that the resulting power distributions satisfy the requirements of the accident analyses.

(continued)

BASES

**APPLICABLE
SAFETY ANALYSES**

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and, to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

The RAOC methodology (Ref. 2) establishes a xenon distribution library with tentatively wide AFD limits. One dimensional axial power distribution calculations are then performed to demonstrate that normal operation power shapes are acceptable for the LOCA and loss of flow accident, and for initial conditions of anticipated transients. The tentative limits are adjusted as necessary to meet the safety analysis requirements.

The limits on the AFD ensure that the Heat Flux Hot Channel Factor ($F_0(Z)$) is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The limits on the AFD also restrict the range of power distributions that are used as initial conditions in the analyses of Condition II, III, or IV events. Compliance with these limits ensures that the fuel cladding integrity is maintained for these postulated accidents. The most important Condition IV event is the LOCA. The most important Condition III event is the complete loss of forced RCS flow accident. The most important Condition II events are uncontrolled bank withdrawal and boration or dilution accidents. Condition II accidents are used to confirm the adequacy of the Overpower ΔT and Overtemperature ΔT trip setpoints.

The limits on the AFD satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The shape of the power profile in the axial (i.e., the vertical) direction is largely under the control of the operator through the manual operation of the control banks or automatic motion of control banks. The automatic motion of the control banks is in response to temperature deviations resulting from manual operation of the Chemical and Volume Control System to change boron concentration or from turbine load changes.

(continued)

BASES

LCO

(continued)

Signals are available to the operator from the Nuclear Instrumentation System (NIS) excore neutron detectors (Ref. 3). Separate signals are taken from the top and bottom power range detectors. The AFD is defined as the difference in normalized flux signals between the top and bottom excore detectors in each detector well. For convenience, this flux difference is converted to provide flux difference units expressed as a percentage and labeled as $\% \Delta$ flux or $\% \Delta I$.

The AFD limits are provided in the COLR. The AFD limits for RAOC do not depend on the target flux difference. However, the target AFD may be used to minimize changes in the axial power distribution.

Violating this LCO on the AFD could produce unacceptable consequences if a Condition II, III, or IV event occurs while the AFD is outside its specified limits.

APPLICABILITY

The AFD requirements are applicable in MODE 1 greater than or equal to 50% RTP when the combination of THERMAL POWER and core peaking factors are of primary importance in safety analysis.

For AFD limits developed using RAOC methodology, the value of the AFD does not affect the limiting accident consequences with THERMAL POWER < 50% RTP and for lower operating power MODES.

ACTIONSA.1

As an alternative to restoring the AFD to within its specified limits, Required Action A.1 requires a THERMAL POWER reduction to < 50% RTP. This places the core in a condition for which the value of the AFD is not important in the applicable safety analyses. A Completion Time of 30 minutes is reasonable, based on operating experience, to reach 50% RTP without challenging plant systems.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.3.1

This Surveillance verifies that the AFD, as indicated by each OPERABLE NIS excore channel, is within its specified limits. The Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. WCAP-8403 (nonproprietary), "Power Distribution Control and Load Following Procedures," Westinghouse Electric Corporation, September 1974.
 2. WCAP-10216-P-A, Revision 1A, Relaxation of Constant Axial Offset Control, F₀ Surveillance Technical Specification, February 1994 (Westinghouse Proprietary).
 3. FSAR, Chapter 15.
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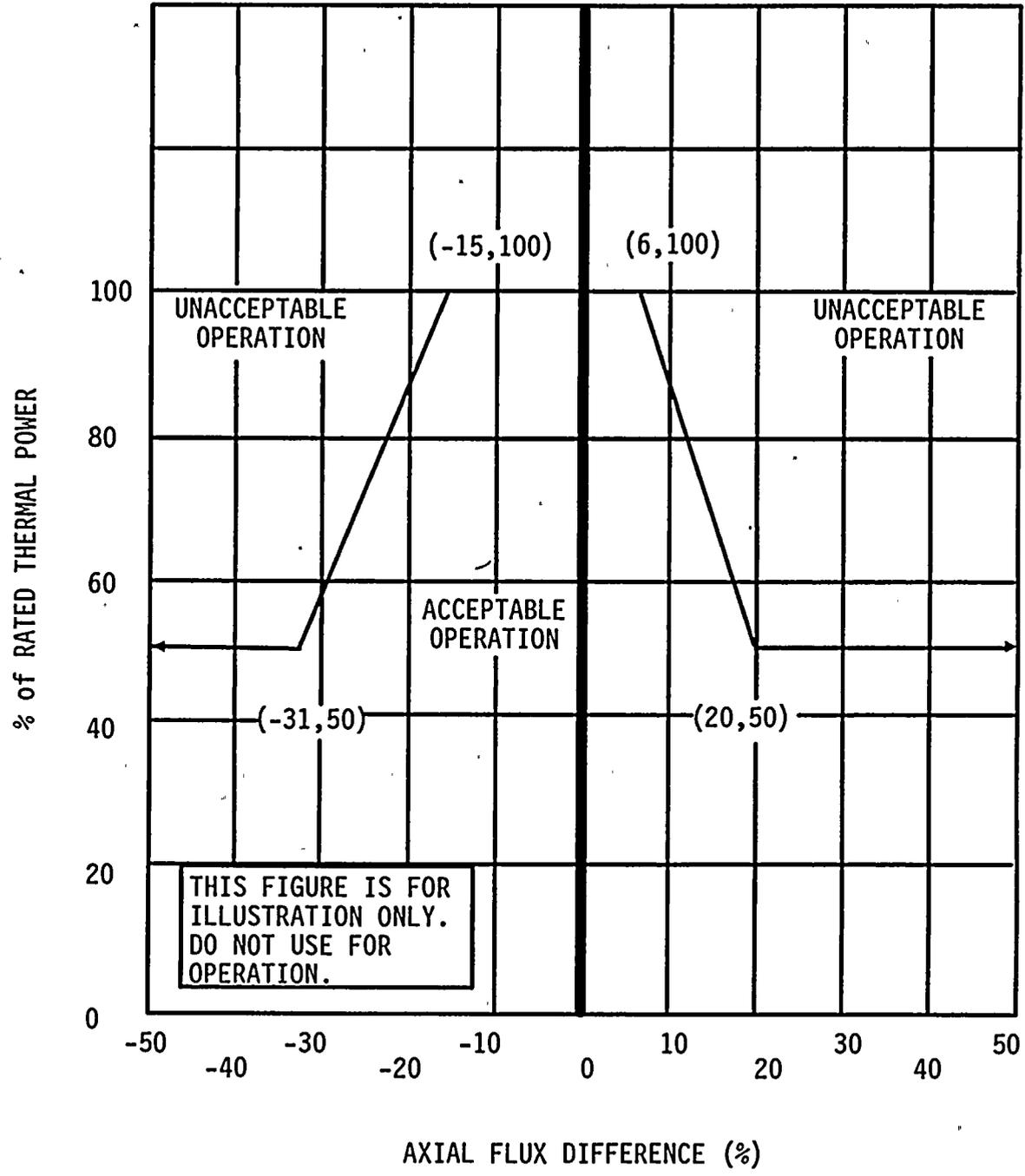


Figure B 3.2.3B-1 (page 1 of 1)
AXIAL FLUX DIFFERENCE Acceptable Operation Limits
as a Function of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE
SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

- a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);
- b. During the Condition 2 partial loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;
- c. During an ejected rod accident, the average fuel pellet enthalpy at the hot spot in irradiated fuel must not exceed 280 cal/gm (Ref. 2); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor (F_{NH}^N), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The QPTR limits ensure that F_{AH}^N and $F_0(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the F_{AH}^N and $F_0(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The QPTR limit of 1.02, above which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_0(Z)$ and (F_{AH}^N) is possibly challenged.

APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the F_{AH}^N and $F_0(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

(continued)

BASES

ACTIONS
(continued)

A.2

After completion of Required Action A.1, the QPTR may still exceed its limits. Any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors F_{AH}^{N} and $F_0(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on F_{AH}^{N} and $F_0(Z)$ within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate F_{AH}^{N} and $F_0(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although F_{AH}^{N} and $F_0(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires

(continued)

BASES

ACTIONS

A.4 (continued)

an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Actions A.1 and A.2, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR remains above the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to eliminate the indicated tilt prior to increasing THERMAL POWER to above the limit of Required Actions A.1 and A.2. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by a Note that states that the indicated tilt is not eliminated until after the evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). This Note is intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the excore detectors are normalized to eliminate the indicated tilt (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_0(Z)$ and F_{alt}^N are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the peaking factor verification cannot be performed within 24 hours due to the non-equilibrium core conditions, a maximum time of 48 hours is allowed for the completion of the verification.

(continued)

BASES

ACTIONS

A.6 (continued)

This Completion Time is intended to allow adequate time to increase THERMAL POWER to above the limit of Required Actions A.1 and A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances must be completed when the excore detectors have been normalized to eliminate the indicated tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which are only required if the excore detectors were normalized to show zero tilt per Required Action A.5.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to $\leq 50\%$ RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is $\leq 75\%$ RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

Input from a Power Range Neutron Flux channel is considered to be operable if the upper and lower detector currents are obtainable. The remaining portion of the channel (the electronics required to provide the channel input to the QPTR alarm) need not be operable.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.2.4.1 (continued)

within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

For those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other power indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channels is inoperable and the THERMAL POWER is > 75% RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one or more power range channels are inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

(continued)

BASES

REFERENCES

1. 10 CFR 50.46.
 2. Regulatory Guide 1.77, Rev [0], May 1974.
 3. 10 CFR 50, Appendix A, GDC 26.
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B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (A00s) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this specification as the Trip Setpoints, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During A00s, which are those events expected to occur more than once during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);
2. Fuel centerline melt shall not occur; and
3. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during A00s.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

(continued)

BASES

BACKGROUND
(continued)

The RTS instrumentation is segmented into four distinct but interconnected modules as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;
2. Signal Process Control and Protection System, including Digital Protection System, Nuclear Instrumentation System (NIS), field contacts, and protection channel sets: provides signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications;
3. Solid State Protection System (SSPS), including input, logic, and output bays: initiates proper unit shutdown and/or ESF actuation in accordance with the defined logic, which is based on the bistable outputs from the signal process control and protection system; and
4. Reactor trip switchgear, including reactor trip breakers (RTBs) and bypass breakers: provides the means to interrupt power to the control rod drive mechanisms (CRDMs) and allows the rod cluster control assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the Trip Setpoint and Allowable Values. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

Signal Process Control and Protection System

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses.

(continued)

BASES

BACKGROUND

Signal Process Control and Protection System (continued)

These setpoints are defined in the FSAR (References 1, 2, 3, 9, 10, & 11). If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the SSPS for decision evaluation, except in the case of the seismic, turbine stop valve position, auto stop oil pressure, 12 kV bus and RCP breaker inputs which do not go through signal conditioning. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, the main control board, the unit computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails, such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. In the case of the Digital Feedwater Control System (DFWCS), the median/signal select (MSS) feature prevents control/protection interaction even though there are only three inputs and 2-out-of-3 logic. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 1.

Two logic channels are required to ensure no single random failure of a logic channel will disable the RTS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing trip.

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

The Trip Setpoints used in the bistables are based on the analytical limits stated in Reference 1. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the WCAP-11082, Rev. 2, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Station - Egel 21 Version," May 1993 (Ref. 6). The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed). Note that in the accompanying LCO 3.3.1, the Trip Setpoints of Table 3.3.1-1 are the LSSS.

Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal, or in the case of the Power Range channels the test signal is added to the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

The Trip Setpoints and Allowable Values listed in Table 3.3.1-1 are based on the methodology described in Reference 6, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

(continued)

BASESBACKGROUND
(continued)Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip and/or ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

The SSPS performs the decision logic for actuating a reactor trip or ESF actuation, generates the electrical output signal that will initiate the required trip or actuation, and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable outputs from the signal processing equipment or relay contact input (RCP breaker, 12kV UV/UF, seismic, etc.) are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various unit transients. If a required logic matrix combination is completed, the system will initiate a reactor trip or send actuation signals via master and slave relays to those components whose aggregate function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Reactor Trip Switchgear

The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity. Each RTB is equipped with a bypass breaker to allow testing of the RTB while the unit is at power. During normal operation the output from the SSPS is a voltage signal that energizes the undervoltage coils in the RTBs and bypass breakers, if in use. When the required logic matrix combination is completed, the SSPS output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTBs and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each reactor trip breaker is also equipped with an automatic shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the SSPS. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

(continued)

BASES

BACKGROUND

Reactor Trip Switchgear (continued)

The decision logic matrix Functions are described in the functional diagrams included in Reference 1. In addition to the reactor trip or ESF, these diagrams also describe the various "permissive interlocks" that are associated with unit conditions. Each train has a built in testing device that can automatically test the decision logic matrix Functions while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

The RTS functions to maintain the applicable limits during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis described in Reference 3 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the accident analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RTS trip Functions that were credited in the accident analysis.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Generally four OPERABLE instrumentation channels in a two-out-of-four configuration are required when one RTS channel is also used as a control system input. In the case of the DFWCS, the MSS feature prevents control/protection interaction even though there are only three inputs and a 2-out-of-3 logic. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with random failure of one of the other three protection channels. Three operable instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RTS trip and disable one RTS channel. The

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY
(continued)

two-out-of-three and two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

Reactor Trip System Functions

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint.

The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel is controlled by a manual reactor trip switch. Each channel activates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE (1-out-of-2 coincidence). These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if the shutdown rods or control rods are withdrawn or the Control Rod Drive (CRD) System is capable of withdrawing the shutdown rods or the control rods. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip does not have to be OPERABLE if the CRD System is not capable of withdrawing the shutdown rods or control rods. If the rods cannot be withdrawn from the core, there is no need to be able to trip the reactor because all of the rods are inserted. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System. Therefore, the actuation logic must be

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2. Power Range Neutron Flux (continued)

able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

a. Power Range Neutron Flux-High

The Power Range Neutron Flux-High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to fuel damage. Reactivity excursions can be caused by rod withdrawal or inadvertent CVCS malfunction, or for example, by sudden changes in RCS coolant temperature such as a feedwater system malfunction (Ref. 12).

The LCO requires all four of the Power Range Neutron Flux-High-channels to be OPERABLE (2-out-of-4 coincidence).

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux-High trip must be OPERABLE. This Function will terminate the reactivity excursion and shutdown the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-High does not have to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

b. Power Range Neutron Flux-Low

The LCO requirement for the Power Range Neutron Flux-Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux-Low channels to be OPERABLE (2-out-of-4 coincidence).

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four

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b. Power Range Neutron Flux-Low (continued)

power range channels are greater than or equal to 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Power Range Neutron Flux Rate

The Power Range Neutron Flux Rate trips use the same channels as discussed for Function 2 above.

a. Power Range Neutron Flux-High Positive Rate

The Power Range Neutron Flux-High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function complements the Power Range Neutron Flux-High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.

The LCO requires all four of the Power Range Neutron Flux-High Positive Rate channels to be OPERABLE (2-out-of-4 coincidence).

In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux-High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed

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a. Power Range Neutron Flux-High Positive Rate
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or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.

b. Power Range Neutron Flux-High Negative Rate

The Power Range Neutron Flux-High Negative Rate trip Function ensures that protection is provided for multiple rod drop accidents. At high power levels, a multiple rod drop accident could cause local flux peaking that would result in an unconservative local DNBR. DNBR is defined as the ratio of the heat flux required to cause a DNB at a particular location in the core to the local heat flux. The DNBR is indicative of the margin to DNB. No credit is taken for the operation of this Function for those rod drop accidents in which the local DNBRs will be greater than the limit.

The LCO requires all four Power Range Neutron Flux-High Negative Rate channels to be OPERABLE (2-out-of-4 coincidence).

In MODE 1 or 2, when there is potential for a multiple rod drop accident to occur, the Power Range Neutron Flux-High Negative Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-High Negative Rate trip Function does not have to be OPERABLE because the core is not critical and DNB is not a concern. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the required SDM is increased during refueling operations. In addition, the NIS power range detectors cannot detect neutron levels present in this MODE.

4. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux-Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons

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4. Intermediate Range Neutron Flux (continued)

leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE (1-out-of-2 coincidence). Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function.

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2 above the P-6 setpoint, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux-High Setpoint trip and the Power Range Neutron Flux-High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range detectors cannot detect neutron levels present in this MODE.

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux-Low Setpoint. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 2 below P-6, 3, 4, and 5 with the Rod Control System capable of rod

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5. Source Range Neutron Flux (continued)

withdrawal or all rods not fully inserted. Therefore, the functional capability at the Trip Setpoint is assumed to be available.

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. The LCO also requires one channel of the Source Range Neutron Flux to be OPERABLE in MODE 3, 4, or 5 with RTBs open or the control rods incapable of withdrawal. In this case, the source range Function is to provide control room indication. The outputs of the Function to RTS logic are not required OPERABLE in MODE 6 or when the RTBs are open or all rods are fully inserted and the Rod Control System is incapable of withdrawal.

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The Function also provides neutron flux indication in the control room.

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE (1-out-of-2 coincidence). Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux-Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range neutron flux trip may be manually blocked and the high voltage to the detectors may be de-energized. Below the P-6 setpoint, the source range neutron flux trip is automatically reinstated and the high voltage to the detectors is automatically energized.

In MODE 3, 4, or 5 with the reactor shut down, but with the Rod Control System capable of rod withdrawal or all rods not fully inserted, the Source Range Neutron Flux trip Function must also be OPERABLE (1-out-of-2 coincidence). If the Rod Control System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like an uncontrolled boron dilution. The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation."

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6. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection and it protects against vessel exit bulk boiling and ensures that the exit quality is within the limits defined by the DNBR correlation. The inputs to the Overtemperature ΔT trip include pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure—the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution— $f(\Delta I)$, the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the
- NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.
- Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

ΔT_0 , as used in the overtemperature and overpower ΔT trips, represents the 100 percent RTP value of ΔT as measured by the plant for each loop. For the initial startup of a refueled core, ΔT_0 is initially assumed to be the same as the last measured ΔT value from the previous cycle until ΔT is measured again at full power. Accurate determination of the loop specific ΔT values should be made quarterly when performing the incore/excore recalibration at steady-state conditions (i.e., power distributions not affected by xenon or other transient conditions). The variation in indicated ΔT between loops is due to the variance in both real hot leg temperatures and hot leg temperature measurement biases.

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6. Overtemperature ΔT (continued)

The real hot leg temperature variance between loops is primarily caused by asymmetrical flow in the upper plenum, and the difference in hot leg temperature measurement biases primarily caused by differences in hot leg temperature streaming error between loops. The change in the indicated loop ΔT s with burn up is caused primarily by the change in the hot leg streaming biases as the radial power distribution changes.

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two loops. The pressure and temperature signals are used for other control functions; thus the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB (2-out-of-4 coincidence). In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions for Condition I and II event (Ref. 12). This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

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7. Overpower ΔT (continued)

- reactor coolant average temperature—the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature—including dynamic compensation for the delays between the core and the temperature measurement system.

ΔT_0 , as used in the overtemperature and overpower ΔT trips, represents the 100 percent RTP value of ΔT as measured by the plant for each loop. For the initial startup of a refueled core, ΔT_0 is initially assumed to be the same as the last measured ΔT value from the previous cycle until ΔT is measured again at full power. Accurate determination of the loop specific ΔT values should be made quarterly when performing the incore/excore recalibration at steady-state conditions (i.e., power distributions not affected by xenon or other transient conditions). The variation in indicated ΔT between loops is due to the variance in both real hot leg temperatures and hot leg temperature measurement biases. The real hot leg temperature variance between loops is primarily caused by asymmetrical flow in the upper plenum, and the difference in hot leg temperature measurement biases is primarily caused by differences in hot leg temperature streaming error between loops. The change in the indicated loop ΔT s with burn up is caused primarily by the change in the hot leg streaming biases as the radial power distribution changes.

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops. The temperature signals are used for other control functions; thus, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the trip setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE (2-out-of-4 coincidence). Note that the Overpower ΔT trip Function receives input from

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7. Overpower ΔT (continued)

consequences of small steamline breaks, as reported in WCAP 9227, Ref. 11, and steamline breaks with coincident control rod withdrawal (Ref. 12). Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure-High and -Low trips and the Overtemperature ΔT trip. The Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System; thus, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

a. Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

The LCO requires four channels of Pressurizer Pressure-Low to be OPERABLE (2-out-of-4 coincidence).

In MODE 1, when DNB is a major concern, the Pressurizer Pressure-Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 Low Pressure Permissive interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent (P-13)). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, there is insufficient heat production to be concerned about DNB.

b. Pressurizer Pressure-High

The Pressurizer Pressure-High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction

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b. Pressurizer Pressure-High (continued)

with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

The LCO requires four channels of the Pressurizer Pressure-High to be OPERABLE (2-out-of-4 coincidence).

The Pressurizer Pressure-High LSSS is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

In MODE 1 or 2, the Pressurizer Pressure-High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure-High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will usually be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Two low temperature overpressure protection system channels provide overpressure protection with the PORVs when below the low temperature cut-off specified in the pressure and temperature limits report (PTLR).

9. Pressurizer Water Level-High

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level-High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

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9. Pressurizer Water Level-High (continued)

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level-High trip must be OPERABLE (2-out-of-3 coincidence). This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions.

10. Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low () trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE (2-out-of-3 coincidence in one loop).

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow-Low trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on low flow in two or more RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

11. Reactor Coolant Pump (RCP) Breaker Position

The RCP Breaker Position trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The position of each RCP breaker is monitored. Above the P-7 setpoint, a loss of flow in two or more loops will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low Trip Setpoint is reached.

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE (2-out-of-4 coincidence). One OPERABLE channel is sufficient for this Function because the RCS

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11. Reactor Coolant Pump (RCP) Breaker Position (continued)

Flow-Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-7 setpoint, the RCP Breaker Position trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in two RCS loops is automatically enabled.

12. Undervoltage Reactor Coolant Pumps

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in all RCS loops. The voltage to both RCP buses is monitored by two relays each. Above the P-7 setpoint, a loss of voltage detected on both RCP buses, i.e. a complete loss of flow event, will initiate a reactor trip. For this event, the under voltage trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires two Undervoltage RCPs channels per bus to be OPERABLE (1-per-bus both busses).

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked, since there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in two-of-four RCS loops is automatically enabled.

13. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. An adequate coastdown time is

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13. Underfrequency Reactor Coolant Pumps (continued)

required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 setpoint, a loss of frequency detected by two relays on one RCP bus will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires two Underfrequency RCPs channels per bus to be OPERABLE (1-per-bus both busses).

In MODE 1 above the P-7 setpoint, the Underfrequency RCPs trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

14. Steam Generator Water Level - Low Low

The SG Water Level - Low Low trip Function ensures that protection is provided against a loss of heat sink in the event of a loss of feedwater flow to one or more SGs. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This Function also performs the ESFAS function of starting the AFW pumps on low low SG level.

The LCO requires three channels of SG Water Level - Low Low per SG (1-per-SG in one SG) and four channels of RCS ΔT (1/loop) to be OPERABLE. The installation of the median signal selector (MSS) and four channels of RCS ΔT (1/loop) effectively eliminates the possibility that a single random failure could cause a control system action that results in a condition requiring protection action, and also prevent proper operation of a protection system channel designed to protect against the condition. Thus, the MSS prevents interaction between the feedwater control and reactor protection systems in accordance with the requirements of

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IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." Removal of this interaction eliminates the need for the low feedwater flow reactor trip. The MSS will functionally separate steam generator narrow range level protection channels (low-low steam generator water level trip) to provide compliance with IEEE 279-1971 and satisfy the original design basis. This trip is actuated on two out of three low-low water level signals occurring in any steam generator. If a low-low water level condition is detected in one steam generator, signals shall be generated to trip the reactor and start the motor driven auxiliary feedwater pumps. If a low-low water level condition is detected in two or more steam generators, a signal is generated to start the turbine driven auxiliary feedwater pump as well.

The signals to actuate reactor trip and start auxiliary feedwater pumps maybe delayed through the use of a Trip Time Delay (TTD) system for reactor power levels below 50% of RTP. Low-low water level in any protection set in any steam generator will generate a signal which starts an elapsed time trip delay timer. The allowable trip time delay is based upon the prevailing power level at the time the low-low level trip setpoint is reached. If power level rises after the trip time delay setpoints have been determined, the trip time delay is re-determined (i.e., decreased) according to the increase in power level. However, the trip time delay is not changed if the power level decreases after the delay has been determined. The use of this delay allows added time for natural steam generator level stabilization or operator intervention to avoid an inadvertent protection system actuation.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level - Low Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level - Low Low Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System in MODE 3 (and 4, prior to going on RHR) and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

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(continued)

15. Steam Generator Water Level - Low, Coincident With Steam Flow/Feedwater Flow Mismatch - Not used.

16. Turbine Trip

a. Turbine Trip - Low Auto Stop Oil Pressure

The Turbine Trip - Low Auto Stop Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, less than or equal to 50% power, will not actuate a reactor trip. Three pressure switches monitor the trip oil pressure in the Turbine Control System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure - High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three channels of Turbine Trip - Low Autostop Oil Pressure to be OPERABLE in MODE 1 above P-9 (2-out-of-3 coincidence).

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip - Low Auto Stop Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip - Turbine Stop Valve Closure

The Turbine Trip - Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, less than or equal to a maximum setpoint of 50 percent power, will not actuate a reactor trip. This trip Function will not and is not

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

b. Turbine Trip-Turbine Stop Valve Closure (continued)

required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip-Low Auto Stop Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated above P-9.

The LSSS for this Function is set to assure channel trip occurs when the associated stop valve is completely closed.

The LCO requires four Turbine Trip-Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. All four channels must trip to cause reactor trip.

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump and Reactor Control Systems. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection, and the Turbine Trip-Stop Valve Closure trip Function does not need to be OPERABLE.

17. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for the small break LOCA but rod insertion is not credited for the large break LOCA (Ref. 3). However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoint and Allowable Values are not applicable to this Function. The SI Input is provided by logic in the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

17. Safety Injection Input from Engineered Safety Feature Actuation System (continued)

SSPS circuitry of ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2 (1-out-of-2 coincidence).

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, . In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

18. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip functions are outside the applicable MODES. These are:

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip and allows the high voltage to be de-energized. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range, and
- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

a. Intermediate Range Neutron Flux, P-6 (continued)

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint (1-out-of-2 coincidence).

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary.

In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection.

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or the Turbine Impulse Pressure, P-13 interlock. The LCO requirement for the P-7 interlock ensures that the following Functions are performed:

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Two Loops);
- RCPs Breaker Open (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

b. Low Power Reactor Trips Block, P-7 (continued)

- (2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:
- Pressurizer Pressure-Low;
 - Pressurizer Water Level-High;
 - Reactor Coolant Flow-Low (Two Loops);
 - RCP Breaker Position (Two Loops);
 - Undervoltage RCPs; and
 - Underfrequency RCPs.

Trip Setpoint and Allowable Value are not applicable to the P-7 interlock because it is a logic Function and thus has no parameter with which to associate an LSSS. The P-7 train is operable if the P-10 and P-13 interlocks are in their required states based on plant conditions.

The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1 (1-out-of-2 coincidence).

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8

The Power Range Neutron Flux, P-8 interlock is actuated at approximately 35% power as determined by two-out-of-four NIS power range detectors. The P-8 interlock automatically enables the Reactor Coolant Flow-Low reactor trips on low flow in one or more RCS loops on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately 35% power. On decreasing

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

c. Power Range Neutron Flux, P-8 (continued)

power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated at less than or equal to 50% power as determined by two-out-of-four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip-Low Auto Stop Oil Pressure and Turbine Trip-Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip may challenge the pressurizer PORVs due to the mismatch between reactor power and the capacities of the Steam Dump and Reactor Control Systems. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

The LCO requires three channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1 (2-out-of-3 coincidence).

In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a significant load rejection

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
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APPLICABILITY
(continued)

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal;
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux-Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a back up signal to block the Source Range Neutron Flux detectors high voltage and allows manual block of the IR rod stop;
- the P-10 interlock provides one of the two inputs to the P-7 interlock; and
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux-Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).
- on decreasing power, the P-10 interlock automatically defeats the block of the source range neutron flux trip and with P-6 energizes the source range high voltage.

The LCO requires three channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2(2-out-of-3).

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a

(continued)

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APPLICABILITY

e. Power Range Neutron Flux, P-10 (continued)

startup or shutdown by the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

f. Turbine Impulse Chamber Pressure, P-13

The Turbine Impulse Chamber Pressure, P-13 interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than 10% of the rated thermal power pressure equivalent. The interlock is determined by one-out-of-two pressure detectors. The LCO requirement for this Function ensures that one of the inputs to the P-7 interlock is available.

The LCO requires two channels of Turbine Impulse Chamber Pressure, P-13 interlock to be OPERABLE in MODE 1 (1-out-of-2-coincidence).

The Turbine Impulse Chamber Pressure, P-13 interlock must be OPERABLE when the turbine generator is operating. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

19. Reactor Trip Breakers

This trip Function applies to the RTBs exclusive of individual trip mechanisms. The LCO requires two OPERABLE trains of trip breakers. A trip breaker train consists of, the trip logic, and all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the Control Rod System. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single random failure can disable the RTS trip capability.

(continued)

BASES

APPLICABLE ,
SAFETY ANALYSES,
LCO, and
APPLICABILITY

19. Reactor Trip Breakers (continued)

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical (1-out-of-2 coincidence). In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Control Rod System is capable of rod withdrawal or all rods are not fully inserted.

20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the Rod Control System, or declared inoperable under Function 19 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical (1-out-of-2 coincidence). In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Rod Control System is capable of rod withdrawal or all rods are not fully inserted.

21. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 19 and 20) and Automatic Trip Logic (Function 21) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is equipped with a bypass breaker to allow testing of the trip breaker while the unit is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE (1-out-of-2 coincidence). Having two OPERABLE channels ensures that random failure of a single logic channel will not prevent reactor trip.

(continued)

BASES

APPLICABLE
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APPLICABILITY

21. Automatic Trip Logic (continued)

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the Control Rod System is capable of rod withdrawal or all rods are not fully inserted.

22. Seismic Trip

The seismic trip system operates to shut down reactor operations should ground accelerations exceed a preset level in any two of the three orthogonal directions monitored (one vertical, two horizontal).

Three triaxial sensors (accelerometers) are anchored to the containment base in three separate locations 120 degrees apart. Each senses acceleration in three mutually orthogonal directions. Output signals are generated when ground accelerations exceed the preset level. These signals are transmitted to the Trains A and B Solid State Protection System (SSPS). If two of the three sensors in any direction produce simultaneous outputs, the logic produces trains A and B reactor trip signals.

The RTS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.1-1 are specified (e.g., on a per steam line, per loop, per basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

(continued)

BASES

ACTIONS
(continued)

A.1

Condition A applies to all RTS protection Functions. Condition A addresses the situation where one or more required channels for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1, B.2.1, and B.2.2

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. This action addresses the train orientation of the SSPS for this Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the safety function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours (54 hours total time). The 6 additional hours to reach MODE 3 are reasonable, based on operating experience, to exit the applicability from full power operation in an orderly manner and without challenging unit systems. With the unit in MODE 3, Condition C is entered if the Manual Reactor Trip Function has not been restored and the Rod Control System is capable of rod withdrawal of all rods are not fully inserted

C.1, C.2.1, and C.2.2

Condition C applies to the following reactor trip Functions in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or all rods not fully inserted:

(continued)

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

- Manual Reactor Trip;
- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

This action addresses the train orientation of the SSPS for these Functions. With one channel or train inoperable, the inoperable channel or train must be restored to OPERABLE status within 48 hours. If the affected Function(s) cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the rods must be fully inserted and the Rod Control System rendered incapable of rod withdrawal within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With the rods fully inserted and the Rod Control System rendered incapable of rod withdrawal, these Functions are no longer required.

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

Condition C is modified by a Note stating that the transition from MODE 5 to MODE 5 with the Rod Control System capable of rod withdrawal or all rods not fully inserted is not permitted for Functions 19, 20, or 21. This Note specifies an exception to LCO 3.0.4 for this MODE 5 transition and avoids placing the plant in a condition where control rods can be withdrawn while the reactor trip system is degraded. This note is in addition to the requirements of LCO 3.0.4 which preclude the transition from either MODE 3 or MODE 4 to MODE 3 or MODE 4 with the Rod control System capable of rod withdrawal or all rods not fully inserted for Functions 19, 20, or 21 with one channel or train inoperable.

D.1.1, D.1.2, D.2.1, D.2.2, and D.3

Condition D applies to the Power Range Neutron Flux-High Function.

The NIS power range detectors provide input to the Rod Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-10271-P-A (Ref. 7).

(continued)

BASES

ACTIONS

D.1.1, D.1.2, D.2.1, D.2.2, and D.3 (continued)

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to $\leq 75\%$ RTP within 12 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design

limits. With one of the NIS power range detectors inoperable, 1/4 of the radial power distribution monitoring capability is lost.

As an alternative to the above actions, the inoperable channel can be placed in the tripped condition within 6 hours and the QPTR monitored once every 12 hours as per SR 3.2.4.2, QPTR verification. Calculating QPTR every 12 hours compensates for the lost monitoring capability due to the inoperable NIS power range channel and allows continued unit operation at power levels $> 75\%$ RTP. The 6 hour Completion Time and the 12 hour Frequency are consistent with LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

As an alternative to the above Actions, the plant must be placed in a MODE where this Function is no longer required OPERABLE. Twelve hours are allowed to place the plant in MODE 3. This is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems. If Required Actions cannot be completed within their allowed Completion Times, LCO 3.0.3 must be entered.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing routine surveillance testing of other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 4 hour time limit is justified in Reference 7.

Required Action D.2.2 has been modified by a Note which states that SR 3.2.4.2 is not required to be performed until 12 hours after input from one Power Range Neutron Flux channel to QPTR becomes inoperable and thermal power is $> 75\%$ RTP. Failure of a component in the Power Range Neutron Flux Channel which renders the High Flux Trip Function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the movable incore detectors once per 12 hours may not be necessary.

(continued)

BASES

ACTIONS
(continued)

E.1 and E.2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux - Low;
- Overtemperature ΔT ;
- Overpower ΔT ;
- Power Range Neutron Flux - High Positive Rate;
- Power Range Neutron Flux - High Negative Rate;
- Pressurizer Pressure - High; and
- SG Water Level - Low Low.

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.

If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The Note also allows placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. The 4 hour time limit is justified in Reference 7.

(continued)

BASESACTIONS
(continued)F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, the overlap of the power range detectors, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

H.1 - Not used

I.1

Condition I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

J.1

Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or all rods not fully inserted. With the unit in this Condition, below P-6, the

(continued)

BASES

ACTIONS

J.1 (continued)

NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition and the unit enters Condition L.

K.1, K.2.1, and K.2.2

Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the Rod Control System capable of rod withdrawal or all rods not fully inserted. With the unit in this Condition, below P-6, the NIS source range performs the protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, 1 additional hour is allowed to fully insert all rods and place the Rod Control System in a condition incapable of rod withdrawal. Once these ACTIONS are completed the core is in a more stable condition. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour to fully insert all rods and place the Rod Control System in a condition incapable of rod withdrawal, are justified in Reference 7.

L.1, L.2, and L.3

Condition L applies when the required number of OPERABLE Source Range Neutron Flux channels is not met in MODE 3, 4, or 5 with the RTBs open or with the Rod Control System incapable of rod withdrawal and all rods fully inserted. With the unit in this Condition, the NIS source range performs a monitoring function. With less than the required number of source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation.

Also, the SDM must be verified within 1 hour and once every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows

(continued)

BASES

ACTIONS

L.1, L.2, and L.3 (continued)

sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Required Action L.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Times of within 1 hour and once per 12 hours are based on operating experience in performing the Required Actions and the knowledge that unit conditions will change slowly.

M.1 and M.2

Condition M applies to the following reactor trip Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low;
- RCP Breaker Position ;
- Undervoltage RCPs; and
- Underfrequency RCPs.

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint (above P-8 for the Reactor Coolant Flow-Low reactor trip function). These Functions do not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below the P-7 setpoint. The 6 hours allowed to place the channel in the tripped condition is justified in Reference 7. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time. The Reactor Coolant Flow - Low reactor trip function does not have to be OPERABLE below the P-8 setpoint; however, the Required Action must take the plant below the P-7 setpoint, if an inoperable channel is not tripped within 6 hours, due to the shared components between this function and the Reactor Coolant Flow - Low trip function.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant

(continued)

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ACTIONS

M.1 and M.2 (continued)

OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition M.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

N.1 and N.2 - Not used

O.1 and O.2 - Not used

P.1 and P.2

Condition P applies to Turbine Trip on Low Auto Stop Oil Pressure or on Turbine Stop Valve Closure. With one or more channels inoperable, the inoperable channel must be placed in the trip condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition. For the Turbine Trip on Turbine Stop Valve Closure function, where four-of-four channels are required to initiate a reactor trip; hence more than one channel may be placed in trip. For the Turbine Trip on Low Auto Stop Oil Pressure function, if one channel is placed in trip, only one additional channel is required to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours. The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 7.

The Required Actions have been modified by a Note that allows placing an inoperable Low Auto Stop Oil Pressure channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.

Q.1 and Q.2

Condition Q applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status (Required Action Q.1) or the unit must be placed in MODE 3 within the

(continued)

BASES

ACTIONS

Q.1 and Q.2 (continued)

next 6 hours. The Completion Time of 6 hours (Required Action Q.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours (Required Action Q.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE.

R.1 and R.2

Condition R applies to the RTBs in MODES 1 and 2. These actions address the train orientation of the RTS for the RTBs. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

The Required Actions have been modified by three Notes. Note 1 allows one RTB to be bypassed for up to 2 hours for surveillance testing or maintenance, provided the other train is OPERABLE. Note 2 allows one RTB to be bypassed only for the time required for performing maintenance on undervoltage or shunt trip mechanisms per Condition U if the other RTB train is OPERABLE. Note 3 allows one RTB to be bypassed for up to 4 hours for logic surveillance testing per Condition Q provided the other train is OPERABLE. The time limits are justified in Reference 7.

S.1 and S.2

Condition S applies to the P-6 and P-10 interlocks. With one or more required channels inoperable, the associated interlock must be verified by observation of the associated permissive annunciator window to be in its required state for the existing unit condition

(continued)

BASES

ACTIONS

S.1 and S.2 (continued)

within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

T.1 and T.2

Condition T applies to the P-7, P-8, P-9, and P-13 interlocks. With one or more required channel(s) inoperable, the associated interlock must be verified by observation of the associated permissive annunciator window to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

U.1 and U.2

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time). The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS

U.1, and U.2 (continued)

With the unit in MODE 3, Condition C is entered if the inoperable trip mechanism has not been restored and the Rod Control System is capable of rod withdrawal or all rods are not fully inserted. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to restore the inoperable trip mechanism to OPERABLE status, consistent with Ref. 13.

The Completion Time of 48 hours for Required Action U.1 is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

V.1 - Not used

W.1

Condition W applies to the Seismic Trip, in MODES 1 and 2. With one of the channels inoperable, START UP and/or POWER OPERATION may proceed provided the inoperable channel is placed in trip within the next 6 hours. If a direction is inoperable, then the channel must be considered inoperable. Placing the channel in the tripped condition creates a partial trip condition requiring only one out of two logic for actuation for that particular location.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 72 hours while performing routine surveillance testing of the other channels. The allowed 72 hour bypass time is reasonable based on the low probability of an event occurring while the channel is bypassed and on the time required to perform the required surveillance testing.

X.1

Condition X applies to the Trip Time Delay (TTD) circuitry for the SG Water Level-Low Low trip function when THERMAL POWER is less than or equal to 50% RTP in MODES 1 and 2. With one or more TTD circuitry delay timers inoperable, adjust the threshold power level for no time delay to 0% RTP, or place the affected SG-low low level in trip. The Completion Time of 6 hours is based on Reference 7.

(continued)

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SURVEILLANCE
REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

SR 3.3.1.1

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS power indications every 24 hours. If the calorimetric exceeds the NIS power indications by > 2% RTP, the NIS is not declared inoperable, but the excore channel gains, must be adjusted consistent with the calorimetric power. If the NIS power indications cannot be properly adjusted, the channel is declared inoperable.

(continued)

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SURVEILLANCE
REQUIREMENTS

SR 3.3.1.2 (continued)

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS power indications shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS power indications and the calorimetric is $> 2\%$ RTP. The second Note clarifies that this Surveillance is required only if reactor power is $\geq 15\%$ RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP but prior to exceeding 30% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is $\geq 3\%$, the NIS channel is still OPERABLE, but must be readjusted.

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the $f(\Delta I)$ input to the overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is $\geq 3\%$. Note 2 clarifies that the Surveillance is required only if reactor power is $\geq 50\%$ RTP and that 24 hours is allowed for performing the first Surveillance after reaching 50% RTP.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, since the changes in neutron flux are slow during the fuel cycle, the expected change in the absolute difference between the incore and excore AFD will be less than 3 percent AFD during this interval.

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SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.4

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and shunt trip mechanisms. Independent verification of RTB undervoltage and shunt trip Function is not required for the bypass breakers. No capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.14. The bypass breaker test shall include a local manual shunt trip only. A Note has been added to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition with the RTB bypass breaker installed, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function including operation of the P-7 permissive which is a logic function only. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the $f(\Delta I)$ input to the overtemperature ΔT Function.

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SURVEILLANCE
REQUIREMENTS

SR 3.3.1.6 (continued)

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 75% RTP and that 24 hours after achieving equilibrium conditions with thermal power $\geq 75\%$ RTP is allowed for performing the first surveillance.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function.

Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

SR 3.3.1.7 is modified by two notes. Note 1 provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3. Note 2 requires that the quarterly COT for the source range instrumentation shall include verification by observation of the associated permissive annunciator window that the P-6 and P-10 interlocks are in their required state for the existing unit conditions.

The Frequency of 92 days is justified in Reference 7.

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BASESSURVEILLANCE
REQUIREMENTS
(continued)SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7 and it is modified by the same Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit conditions by observation of the associated permissive annunciator window. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 92 days of the Frequencies prior to reactor startup, 12 hours after reducing power below P-10, and four hours after reducing power below P-6, as discussed below. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of "12 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup, 12 hours after reducing power below P-10, and four hours after reducing power below P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 for more than 12 hours or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 12 hour or 4 hour limit, as applicable. These time limits are reasonable, based on operating experience, to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for the periods discussed above.

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every 92 days, as justified in Reference 7.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.9 (continued)

The SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to RCP undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION.

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the DCPD setpoint methodology.

The Frequency of 18 months is based on the assumed calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This SR is modified by three Notes. Note 1 states that neutron detectors are excluded from the CHANNEL CALIBRATION. Note 2 states that the test shall include verification that the time constants are adjusted to the prescribed values where applicable. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP. The CHANNEL CALIBRATION for the source range neutron detectors consists of obtaining the detector

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.11 (continued)

plateau curves, evaluating those curves, and comparing the curves to the manufacturer's data. For the intermediate range and power range channels, a test shall be performed that shows allowed variances of detector voltage do not effect detector operation. This SR is also modified by Note 3 stating that this surveillance is not required to be performed until reactor power exceeds P-6 because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The source range plateau curves are obtained under the conditions that apply during a plant outage.

The 18 month Frequency is based on past operating experience, which has shown these components usually pass the Surveillance when performed on the 18 month Frequency. The conditions for obtaining the source range plateau curves and the power and intermediate range detector voltages are described above. The other remaining portions of the CHANNEL CALIBRATIONS may be performed either during a plant outage or during plant operation.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a CHANNEL CALIBRATION of the seismic trip, every 18 months.

CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.13

SR 3.3.1.13 is the performance of a COT of RTS interlocks every 18 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, Seismic Trip and the SI Input from ESFAS. This TADOT is performed every 18 months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.15

SR 3.3.1.15 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to reactor startup. A Note states that this Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.

SR 3.3.1.16

SR 3.3.1.16 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in the FSAR (Ref. 1). Individual component response times are not modeled in the analyses.

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SURVEILLANCE
REQUIREMENTSSR 3.3.1.16 (continued)

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

The response time testing for the SG water level low-low does not include trip time delays. Response times include the transmitters, Eagle-21 process protection cabinets, solid state protection system cabinets, and actuation devices only. This reflects the response times necessary for THERMAL POWER in excess of 50 percent RTP. For those functions without a specified response time, SR 3.3.1.16 is not applicable.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in-place, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" (Ref. 8) provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocations for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.1.16 (continued)

As appropriate, each channel's response time must be verified every 18 months on a STAGGERED TEST BASIS. Each verification shall include at least one train such that both trains are verified at least once per 36 months. Testing of the final actuation devices is included in the verification. Response times cannot be determined during unit operation because equipment operation is required to measure response times. Experience has shown that these components usually pass this surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.16 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input to the first electronic component in the channel.

SR 3.3.1.17

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST for the Seismic trip. The frequency of every 18 months is based on instrument reliability and operating history data.

REFERENCES

1. FSAR, Chapter 7.
2. FSAR, Chapter 6.
3. FSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. WCAP-11082, Rev. 2, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Station - Eage1 21 Version." May 1993
7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
8. WCAP 13632 - PA-1, Rev. 2 "Elimination of Pressure Sensor Response Time Testing Requirements."
9. FSAR, Chapter 9.2.7 & 9.2.2.

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10. FSAR, Chapter 10.3 & 10.4.
 11. FSAR, Chapter 8.3.
 12. DCM S-38A, "Plant Protection System"
 13. WCAP-13878, "Reliability of Potter & Brumfield MDR Relays",
June 1994.
 14. WCAP-13900, "Extension of Slave Relay Surveillance Test
intervals", April 1994.
 15. WCAP-14117, "Reliability Assessment of Potter and Brumfield
MDR Series Relays."
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B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured;
- Signal processing equipment including digital protection system, field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications; and
- Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable outputs from the signal process control and protection system.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). In some cases, the same channels also provide control system inputs. To account for calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the Trip Setpoint and Allowable

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BASES

BACKGROUND Field Transmitters or Sensors (continued)

Values. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

Signal Processing Equipment

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in FSAR, Chapter 6 (Ref. 1), Chapter 7 (Ref. 2), and Chapter 15 (Ref. 3). If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the SSPS for decision evaluation. Channel separation is maintained up to and through the input bays. However, not all unit parameters require four channels of sensor measurement and signal processing. Some unit parameters provide input only to the SSPS, while others provide input to the SSPS, the main control board, the unit computer, and one or more control systems.

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the SSPS and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. In the case of the Digital Feedwater Control System (DFWCS), the median/signal select (MSS) feature prevents control/protection interaction even though there are only three inputs and 2-out-of-3 logic. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function

(continued)

BASES

BACKGROUND Signal Processing Equipment (continued)

actuation. Again, a single failure will neither cause nor prevent the protection function actuation.

These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 2.

The channels are designed such that testing required to be performed at power may be accomplished without causing an ESF actuation.

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

The Trip Setpoints used in the bistables are based on the analytical limits stated in Reference 2. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.2-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the WCAP-11082, Rev. 2, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Station - Eagle 21 Version," May 1993(Ref. 6). The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

(continued)

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BACKGROUND

Trip Setpoints and Allowable Values (continued)

Certain channels can be tested on line to verify that the signal processing equipment and setpoint accuracy is within the specified allowance requirements of Reference 2. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SR section.

The Trip Setpoints and Allowable Values listed in Table 3.3.2-1 are based on the methodology described in Reference 6, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements.

The SSPS performs the decision logic for most ESF equipment actuation; generates the electrical output signals that initiate the required actuation; and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable outputs from the signal processing equipment are sensed by the SSPS equipment and combined into logic matrices that represent combinations indicative of various

(continued)

BASES

BACKGROUND

Solid State Protection System (continued)

transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Each SSPS train has a built in testing device that can automatically test the decision logic matrix functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays, which then cause actuation of the end-devices. The master and slave relays are routinely tested to ensure operation. The test of the master relays energizes the relay, which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The SLAVE RELAY TEST actuates the devices if their operation will not interfere with continued unit operation. For the latter case, actual component operation is prevented by the SLAVE RELAY TEST circuit, and slave relay contact operation is verified by a continuity check of the circuit containing the slave relay. The SLAVE RELAY TEST is performed on a refueling frequency. The test frequency is based on relay reliability assessments presented in WCAP-13878, "Reliability Assessment of Potter and Brumfield MDR Series Relays," WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," and WCAP-14117, "Reliability Assessment of Potter and Brumfield MDR Series Relay." These reliability assessments are relay specific and apply only to Potter and Brumfield MDR series relays which are the only relays used in the ESF actuation system. Note that for normally energized applications, the relays may have to be replaced periodically in accordance with the guidance given in WCAP-13878 for MDR relays.

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Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure-Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3).

The LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped or bypassed during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to < 2200°F); and

(continued)

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1. Safety Injection (continued)
2. Boration to ensure recovery and maintenance of SDM ($k_{\text{eff}} < 1.0$).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other functions such as:

- Phase A Isolation;
- Containment Ventilation Isolation;
- Reactor Trip;
- Turbine Trip from Reactor Trip with P-9;
- Feedwater Isolation and Feedwater Pump Turbine Trip;
- Start of motor driven auxiliary feedwater (AFW) pumps;
- Control room ventilation to pressurization mode, and Auxiliary Building to Building and Safeguards or safeguards only mode;
- Start the diesel generators (DGs) and transfer to the startup bus;
- Start the containment fan cooler units (CFCUs) in low speed;
- Start the component cooling water and auxiliary salt water pumps;
- Input to containment spray pump and discharge valve auto start (with containment spray signal);
- Isolate SG sample blowdown lines.

These other functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the turbine and reactor to limit power generation;
- Isolation of main feedwater (MFW) to limit secondary side mass losses;

(continued)

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1. Safety Injection (continued)

- Start of AFW to ensure secondary side cooling capability;
- Start the DGs to compensate for a possible loss of offsite power (LOOP);
- Start the components associated with the accident heat removal systems.

a. Safety Injection—Manual Initiation

The LCO requires one channel per train to be OPERABLE. The operator can initiate SI at any time by using either of two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained in the manual ESFAS actuation circuitry to ensure the operator has manual ESFAS initiation capability.

Each channel consists of one control switch and the interconnecting wiring to the actuation logic cabinet. Each control switch actuates both trains. This configuration does not allow testing at power.

b. Safety Injection—Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Manual and automatic initiation of SI must be OPERABLE in MODES 1, 2, and 3. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Manual Initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but

(continued)

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b. Safety Injection - Automatic Actuation Logic and Actuation Relays (continued)

because of the large number of components actuated on a SI, actuation is simplified by the use of the manual actuation control switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation.

These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Unit pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

c. Safety Injection - Containment Pressure - High

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. The transmitters (d/p cells) and electronics are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment.

Thus, the high pressure Function will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

(continued)

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c. Safety Injection - Containment Pressure - High
(continued)

Containment Pressure - High must be OPERABLE in MODES 1, 2, 3, and 4 when there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 5, and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment.

d. Safety Injection - Pressurizer Pressure - Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) relief or safety valve;
- SLB;
- A spectrum of rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer relief or safety valve;
- LOCAs; and
- SG Tube Rupture.

Pressurizer pressure provides both control and protection functions: input to the Pressurizer Pressure Control System, reactor trip, and SI. Therefore, the actuation logic must be able to withstand both an input failure to control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with a two-out-of-four logic.

(continued)

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d. Safety Injection - Pressurizer Pressure - Low
(continued)

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above P-11) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the P-11 setpoint. Automatic SI actuation below this pressure setpoint is then performed by the Containment Pressure - High signal.

This Function is not required to be OPERABLE in MODE 3 below the P-11 setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

e. Safety Injection - Steam Line Pressure

(1) Steam Line Pressure - Low

Steam Line Pressure - Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG relief or an SG safety valve.

Steam Line Pressure - Low provides input to the DFWCS functions. The MSS function prevents the excursion of one of the inputs from causing a process disturbance that would require protective action from the remaining channels on the affected steam line. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective

(continued)

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(1) Steam Line Pressure - Low (continued)

requirements with a two-out-of-three logic on each steam line.

With some transmitters located inside the penetration area, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

This Function is anticipatory in nature and has a lead/lag ratio of 50/5.

Steam Line Pressure - Low must be OPERABLE in MODES 1, 2, and 3 (above P-11) when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, feed line break is not a concern. Inside containment SLB will be terminated by automatic SI actuation via Containment Pressure - High, and outside containment SLB will be terminated by the Steam Line Pressure - Negative Rate - High signal for steam line isolation. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to cause an accident.

(2) Not used

f, g. Not used

2. Containment Spray

Containment Spray coincident with an SI signal provides three primary functions:

1. Lowers containment pressure and temperature after an HELB in containment;
2. Reduces the amount of radioactive iodine in the containment atmosphere; and
3. Adjusts the pH of the water in the containment recirculation sump after a large break LOCA.

These functions are necessary to:

(continued)

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2. Containment Spray (continued)

- Ensure the pressure boundary integrity of the containment structure;
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure; and
- Minimize corrosion of the components and systems inside containment following a LOCA.

The containment spray actuation signal coincident with SI starts the containment spray pumps and aligns the discharge of the pumps to the containment spray nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the containment spray pumps and mixed with a sodium hydroxide solution from the spray additive tank. When the RWST reaches the low low level setpoint, the spray pumps are manually tripped. Spray flow can then be shifted to the RHR system if continued containment spray is required. Containment spray is actuated by or Containment Pressure-High High coincident with an SI signal.

a. Containment Spray- Manual Initiation

The operator can manually initiate containment spray from the control room if an SI signal is present by simultaneously turning both Containment Isolation Phase "B"/containment spray Actuate Trains A & B switches. Because an inadvertent actuation of containment spray could have such serious consequences, two switches must be turned

(continued)

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a. Containment Spray—Manual Initiation (continued)

simultaneously and an SI signal must be present to initiate APPLICABILITY containment spray. Simultaneously turning the two switches in either set will actuate containment spray in both trains in the same manner as the automatic actuation signal. Two Manual Initiation switches in each train are required to be OPERABLE. Note that Manual Initiation of containment spray also actuates Phase B containment isolation and CVI.

b. Containment Spray—Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of containment spray must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for an accident to occur, and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. Manual initiation is also required in MODE 4, even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

(continued)

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(continued)

c. Containment Spray - Containment Pressure

This signal provides protection against a LOCA or an SLB inside containment. The transmitters (d/p cells) are located outside of containment with the sensing line (high pressure side of the transmitter) located inside containment. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

This is one of the Functions that requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate containment spray, since the consequences of an inadvertent actuation of containment spray could be serious. Note that this Function also has the inoperable channel placed in bypass rather than trip to decrease the probability of an inadvertent actuation.

The logic configuration is a two-out-of-four. This configuration is called the Containment Pressure - High High Setpoint. Additional redundancy is warranted because this Function is energize to trip. Containment Pressure - High High must be OPERABLE in MODES 1, 2, 3, and 4 when there is sufficient energy in the primary and secondary sides to pressurize the containment following a pipe break. In MODES 5, and 6, there is insufficient energy in the primary and secondary

(continued)

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c. Containment Spray - Containment Pressure
(continued)

sides to pressurize the containment and reach the
Containment Pressure - High High setpoints.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except component cooling water (CCW), at a relatively low containment pressure indicative of primary or secondary system leaks. For these types of events, forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since CCW is required to support RCP operation, not isolating CCW on the low pressure Phase A signal enhances unit safety by allowing operators to use forced RCS circulation to cool the unit. Isolating CCW on the low pressure signal may force the use of feed and bleed cooling, which could prove more difficult to control.

Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation logic. All process lines penetrating containment, with the exception of CCW, are isolated. CCW is not isolated at this time to permit continued operation of the RCPs with cooling water flow to the thermal barrier heat exchangers and oil coolers. All process lines not equipped with remote operated isolation valves are manually closed, or otherwise isolated, prior to reaching MODE 4.

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains. Note that manual

(continued)

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3. Containment Isolation (continued)

actuation of Phase A Containment Isolation also actuates Containment Ventilation Isolation.

The Phase B signal isolates CCW. This occurs at a relatively high containment pressure that is indicative of a large break LOCA or an SLB. For these events, forced circulation using the RCPs is no longer desirable. Isolating the CCW at the higher pressure does not pose a challenge to the containment boundary because the CCW System is a closed loop inside containment. Although some system components do not meet all of the ASME Code requirements applied to the containment itself, the system is continuously pressurized to a pressure greater than the Phase B setpoint. Thus, routine operation demonstrates the integrity of the system pressure boundary for pressures exceeding the Phase B setpoint. Furthermore, because system pressure exceeds the Phase B setpoint, any system leakage prior to initiation of Phase B isolation would be into containment. Therefore, the combination of CCW System design and Phase B isolation ensures the CCW System is not a potential path for radioactive release from containment.

Phase B containment isolation is actuated by Containment Pressure-High High, or manually, via the automatic actuation logic, as previously discussed. For containment pressure to reach a value high enough to actuate Containment Pressure-High High, a large break LOCA or SLB must have occurred and containment spray must have been actuated. RCP operation will no longer be required and CCW to the RCPs is, therefore, no longer necessary. The RCPs can be operated with seal injection flow alone and without CCW flow to the thermal barrier heat exchanger.

Manual Phase B Containment Isolation is accomplished by the same switches that actuate Containment Spray. When the two switches are operated simultaneously, Phase B Containment Isolation and Containment Spray will be actuated in both trains.

(continued)

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(continued)

a. Containment Isolation - Phase A Isolation

(1) Phase A Isolation - Manual Initiation

Manual Phase A Containment Isolation is actuated by either of two switches in the control room. Either switch actuates both trains. Note that manual initiation of Phase A Containment Isolation also actuates Containment Ventilation Isolation.

(2) Phase A Isolation - Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of Phase A Containment Isolation must be OPERABLE in MODES 1, 2, 3, and 4, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a Phase A Containment Isolation, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase A Containment Isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase A Isolation - Safety Injection

Phase A Containment Isolation is also initiated by all Functions that initiate SI. The Phase A Containment Isolation

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(3) Phase A Isolation-Safety Injection
(continued)

requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

b. Containment Isolation-Phase B Isolation

Phase B Containment Isolation is accomplished by Manual Initiation, Automatic Actuation Logic and Actuation Relays, and by Containment Pressure channels (the same channels that actuate Containment Spray, Function 2). The Containment Pressure trip of Phase B Containment Isolation is energized to trip in order to minimize the potential of spurious trips that may damage the RCPs.

(1) Phase B Isolation-Manual Initiation

(2) Phase B Isolation-Automatic Actuation Logic and Actuation Relays

Manual and automatic initiation of Phase B containment isolation must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for an accident to occur. Because of the large number of components actuated on a Phase B containment isolation, actuation is simplified by the use of the manual actuation switches. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase B containment

(continued)

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- (1) Phase B Isolation - Manual Initiation
- (2) Phase B Isolation - Automatic Actuation Logic and Actuation Relays (continued)

isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

- (3) Phase B Isolation - Containment Pressure

The basis for containment pressure MODE applicability is as discussed for ESFAS Function 2.c above.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For an SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize.

a. Steam Line Isolation - Manual Initiation

Manual initiation of Steam Line Isolation can be accomplished from the control room via an individual switch on each valve. The LCO requires one channel per valve to be OPERABLE.

(continued)

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(continued)

b. Steam Line Isolation - Automatic Actuation Logic
and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation - Containment Pressure - High High

This Function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment to limit the mass and energy release to containment to a single SG. The transmitters (d/p cells) are located outside containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure - High - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. However, for enhanced reliability, this Function was designed with four channels and a two-out-of-four logic. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions, and the Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure - High - High must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe

(continued)

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c. Steam Line Isolation-Containment Pressure-High 2
(continued)

break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed and de-activated, i.e., actions are taken to assure that the valve cannot be inadvertently opened. In MODE 4, the increase in containment pressure following a pipe break would occur over a relatively long time period such that manual action could reasonable be expected to provide protection and ESFAS Function 4.d need not be OPERABLE. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure-High -High setpoint.

d. Steam Line Isolation- Steam Line Pressure

(1) Steam Line Pressure-Low

Steam Line Pressure-Low provides closure of the MSIVs in the event of an SLB to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. This function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven AFW pump. Steam Line Pressure-Low was discussed previously under SI Function 1.e.1.

Steam Line Pressure-Low Function must be OPERABLE in MODES 1, 2, and 3 (above P-11), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure-High -High. Stuck valve transients and outside containment SLBs will be terminated by the Steam Line Pressure-Negative Rate-High signal for Steam Line Isolation below P-11 when SI has been manually blocked. The Steam Line Isolation Function is required in MODES 2

(continued)

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(1) Steam Line Pressure - Low (continued)

and 3 unless all MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

(2) Steam Line Pressure - Negative Rate - High

Steam Line Pressure - Negative Rate - High provides closure of the MSIVs for an SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure - Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure - Negative Rate - High signal is automatically enabled. Steam Line Pressure - Negative Rate - High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy requirements with a two-out-of-three logic on each steam line.

Steam Line Pressure - Negative Rate - High must be OPERABLE in MODE 3 (2 per steam line) when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure - Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

(2) Steam Line Pressure - Negative Rate - High
(continued)

While the transmitters may experience elevated ambient temperatures due to an SLB, the trip function is based on rate of change, not the absolute accuracy of the indicated steam pressure. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

e, f. Not used

g. Not used

h. Not used

5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, and to stop the excessive flow of feedwater into the SGs. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

The Function is actuated when the level in any SG exceeds the high high setpoint, and performs the following functions:

- Trips the main turbine;
- Trips the MFW pumps;
- Initiates feedwater isolation; and
- Shuts the MFW regulating valves and the bypass feedwater regulating valves coincident with P-4.

This Function is actuated by SG Water Level - High High. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

5. Turbine Trip and Feedwater Isolation (continued)

a. Turbine Trip and Feedwater Isolation - Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Turbine Trip and Feedwater Isolation - Steam Generator Water Level - High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments provide input to the SG Water Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system (which may then require the protection function actuation) and a single failure in the other channels providing the protection function actuation. Thus, three OPERABLE channels (narrow range instrument span each generator) are required to satisfy the requirements with a two-out-of-three logic and a median signal selector is provided to prevent control and protection function interactions.

The transmitters (d/p cells) are located inside containment. However, the events that this Function protects against cannot cause a severe environment in containment. Therefore, the Trip Setpoint reflects only steady state instrument uncertainties.

c. Turbine Trip and Feedwater Isolation - Safety Injection

Turbine Trip and Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

Turbine Trip and Feedwater Isolation Functions must be OPERABLE in MODES 1 and 2 except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve when the MFW System is in operation and the turbine generator may be in operation. In MODES 3, 4, 5, and 6, the MFW System and the turbine

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

c. Turbine Trip and Feedwater Isolation - Safety Injection (continued)

generator are not in service and this Function is not required to be OPERABLE.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The system has two motor driven pumps and a turbine driven pump, making it available during normal unit operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break. The normal source of water for the AFW System is the condensate storage tank (CST). The AFW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately.

a. Auxiliary Feedwater - Manual Initiation

Manual initiation of Auxiliary Feedwater can be accomplished from the Control Room. Each of the three AFW pumps has a switch for manual initiation. The LCO requires three channels to be OPERABLE.

b. Auxiliary Feedwater - Automatic Actuation Logic and Actuation Relays (Solid State Protection System)

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

c. Not used

d. Auxiliary Feedwater - Steam Generator Water Level - Low Low

SG Water Level - Low Low provides protection against a loss of heat sink. A feed line break, inside or outside of containment, or a loss of MFW, would result in a loss of SG water level. SG Water Level - Low Low provides input to the SG

(continued)

BASES

APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

d. Auxiliary Feedwater - Steam Generator Water
Level - Low Low (continued)

Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system which may then require a protection function actuation and a single failure in the other channels providing the protection function actuation. Thus, three OPERABLE channels (narrow range instrument span each generator) are required to satisfy the requirements with two-out-of-three logic and a median signal selector is provided for level control.

This function is actuated on two out of three low-low water level signals occurring in any steam generator. If a low-low water level condition is detected in one steam generator, signals are generated to start the motor driven auxiliary feedwater pumps. If a low-low water level condition is detected in two or more steam generators, a signal is generated to start the turbine driven auxiliary feedwater pump as well.

The signals to start auxiliary feedwater pumps are delayed through the use of a Trip Time Delay (TTD) system for reactor power levels below 50% of RTP. Low-low water level in any protection set in any steam generator will generate a signal which starts an elapsed time trip delay timer. The allowable trip time delay is based upon the prevailing power level at the time the low-low level trip setpoint is reached. If power level rises after the trip time delay setpoints have been determined, the trip time delay is re-determined (i.e., decreased) according to the increase in power level. However, the trip time delay is not changed if the power level decreases after the delay has been determined. The use of this delay allows added time for natural steam generator level stabilization or operator intervention to avoid an inadvertent protection system actuation.

With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

e. Auxiliary Feedwater - Safety Injection

An SI signal starts the motor driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1.

(continued)

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APPLICABLE
SAFETY ANALYSES,
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APPLICABILITY

e. Auxiliary Feedwater - Safety Injection (continued)

Instead, Function 1, SI, is referenced for all initiating functions and requirements.

f. Not used

Functions 6.a, 6.b, 6.d, and 6.e must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor except the RCS ΔT time delays associated with Function 6.d, are only required to be operable in MODES 1 and 2. Below Mode 2, for the trip time delay, the maximum time delay is permitted; therefore, no OPERABILITY requirement is imposed on vessel ΔT channels in MODE 3. SG Water Level - Low Low in any operating SG will cause the motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level - Low Low in any two operating SGs will cause the turbine driven pumps to start. These Functions do not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

g. Auxiliary Feedwater - Undervoltage Reactor Coolant Pump

A loss of power on the buses that provide power to the RCPs provides indication of a pending loss of RCP forced flow in the RCS. The Undervoltage RCP Function senses the voltage upstream of each RCP breaker. A loss of power, on two RCPs buses, will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip.

h. Not used

Function 6.g must be OPERABLE in MODE 1. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 2, 3, 4, and 5, the pump

(continued)

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(continued)

trip is not indicative of a condition requiring automatic AFW initiation of the TDAFW pump. No other anticipatory start signals are necessary for the TDAFW pump, only low level in 2 of 4 SGs.

i. Not used

7. Not used

8. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

a. Engineered Safety Feature Actuation System Interlocks - Reactor Trip, P-4

The P-4 interlock is enabled when a reactor trip breaker (RTB) and its associated bypass breaker is open. This Function allows operators to manually block reactivation of SI systems after the initial phase of injection is complete. Once SI is blocked, automatic actuation of SI cannot occur until the RTBs have been manually closed. The functions of the P-4 interlock are:

- Trip the main turbine;
- Isolate MFW with coincident low $T_{avg} \leq 554^{\circ}\text{F}$;
- Allows manual block of the automatic reactivation of SI ;
- Transfer the steam dump from the load rejection controller to the plant trip controller; and
- Prevent opening of the MFW reg valves or bypass valves if they were closed on SI or high SG Water Level.

Each of the above Functions is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the

(continued)

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APPLICABLE
SAFETY ANALYSES,
LCO, and
APPLICABILITY

a. Engineered Safety Feature Actuation System
Interlocks - Reactor Trip, P-4 (continued)

RCS following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in core power. To avoid such a situation, the noted Functions have been interlocked with P-4 as part of the design of the unit control and protection system.

None of the noted Functions serves a mitigation function in the unit licensing basis safety analyses. Only the turbine trip Function is explicitly assumed since it is an immediate consequence of the reactor trip Function. Neither turbine trip, nor any of the other four Functions associated with the reactor trip signal, is required to show that the unit licensing basis safety analysis acceptance criteria are met.

The RTB position switches that provide input to the P-4 interlock only function to energize or de-energize or open or close contacts. Therefore, this Function has no adjustable trip setpoint with which to associate a Trip Setpoint and Allowable Value.

This Function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical or approaching criticality. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because the main turbine, the MFW System, and the Steam Dump System are not in operation.

b. Engineered Safety Feature Actuation System
Interlocks - Pressurizer Pressure, P-11

The P-11 interlock permits a normal unit cooldown and depressurization without actuation of SI or main steam line isolation. With two-out-of-three pressurizer pressure channels (discussed previously) less than the P-11 setpoint, the operator can manually block the Pressurizer Pressure-Low and Steam Line Pressure-Low SI signals and the Steam Line Pressure-Low steam line isolation signal (previously discussed). When the Steam Line Pressure-Low steam line isolation signal is manually blocked, a main steam isolation signal on Steam Line Pressure-Negative Rate-High is enabled. This provides protection for an SLB by closure of the MSIVs. With two-out-of-three pressurizer pressure channels above the P-11 setpoint, the Pressurizer Pressure-Low and Steam Line Pressure-Low SI signals and the Steam Line Pressure-Low steam line isolation

(continued)

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APPLICABLE
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LCO, and
APPLICABILITY

b. Engineered Safety Feature Actuation System
Interlocks - Pressurizer Pressure, P-11 (continued)

signal are automatically enabled. The operator can also enable these trips by use of the respective manual reset switches. When the Steam Line Pressure-Low steam line isolation signal is enabled, the main steam isolation on Steam Line Pressure-Negative Rate-High is disabled. The Trip Setpoint reflects only steady state instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 to allow an orderly cooldown and depressurization of the unit without the actuation of SI or main steam isolation. This Function does not have to be OPERABLE in MODE 4,

5, or 6 because system pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

c. Not used ✓

The ESFAS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.2-1 are specified (e.g., on a per steam line, per loop, per SG, etc., basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

When the number of inoperable channels in a trip function exceed those specified in one or other related Conditions associated with a trip function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

(continued)

BASES

ACTIONS
(continued)

A.1

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one or more channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1, B.2.1 and B.2.2

Condition B applies to manual initiation of:

- SI;
- Containment Spray;
- Phase A Isolation; and
- Phase B Isolation.

This action addresses the train orientation of the SSPS for the functions listed above. If a channel or train is inoperable, 48 hours is allowed to return it to an OPERABLE status. Note that for containment spray and Phase B isolation, failure of one or both channels in one train renders the train inoperable. Condition B, therefore, encompasses both situations. The specified Completion Time is reasonable considering that there are two automatic actuation trains and another manual initiation train OPERABLE for each Function, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (54 hours total time) and in MODE 5 within an additional 30 hours (84 hours total time). The allowable Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1 and C.2.2

Condition C applies to the automatic actuation logic and actuation relays for the following functions:

- SI;
- Containment Spray;

(continued)

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ACTIONS

C.1, C.2.1 and C.2.2 (continued)

- Phase A Isolation;
- Phase B Isolation

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (12 hours total time) and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to [4] hours for surveillance testing, provided the other train is OPERABLE. This allowance is based on the reliability analysis assumption of WCAP-10271-P-A (Ref. 8) that 4 hours is the average time required to perform channel surveillance.

D.1, D.2.1, and D.2.2

Condition D applies to:

- Pressurizer Pressure - Low ;
- Steam Line Pressure - Low;
- Containment Pressure - High - High
- Steam Line Pressure - Negative Rate - High;
- Steam Line Pressure - Low;
- Flow in Two Steam Lines Coincident ;
- SG Water level - Low Low; .

If one channel is inoperable, 6 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic (excluding pressurizer pressure - low and containment pressure high - high). Therefore, failure of one channel places the Function in a two-out-of-two configuration. The inoperable channel must be tripped to place

(continued)

BASES

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

the Function in a one-out-of-two configuration that satisfies redundancy requirements.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel or one additional channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 4 hours allowed for testing, are justified in Reference 8.

E.1, E.2.1, and E.2.2

Condition E applies to:

- Containment Pressure - High;
- Containment Spray Containment Pressure High, High ; and
- Containment Phase B Isolation Containment Pressure- (High, High).

None of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray. The containment spray signal is also interlocked with SI and will not initiate without simultaneous SI and case spray signals.

(continued)

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ACTIONS

E.1, E.2.1, and E.2.2 (continued)

To avoid the inadvertent actuation of containment spray and Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 6 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status, or place it in the bypassed condition within 6 hours, requires the unit be placed in MODE 3 within the following 6 hours and MODE 5 within 42 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows one additional channel to be bypassed for up to 4 hours for surveillance testing. Placing a second channel in the bypass condition for up to 4 hours for testing purposes is acceptable based on the results of Reference 8.

F.1, F.2.1, and F.2.2

Condition F applies to the P-4 Interlock.

For the P-4 Interlock Function, this action addresses the train orientation of the SSPS. If a train is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time is reasonable considering the nature of this Function, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above.

(continued)

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ACTIONS
(continued)

G.1, G.2.1 and G.2.2

Condition G applies to the automatic actuation logic and actuation relays for the Steam Line Isolation and AFW actuation Functions.

The action addresses the train orientation of the SSPS and the master and slave relays for these functions. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to [4] hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 8) assumption that 4 hours is the average time required to perform channel surveillance.

H.1 and H.2

Condition H applies to the automatic actuation logic and actuation relays for the Turbine Trip and Feedwater Isolation Function.

This action addresses the train orientation of the SSPS and the master and slave relays for this Function. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the following 6 hours. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of

(continued)

BASES

ACTIONS

H.1 and H.2 (continued)

an event occurring during this interval. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. These Functions are no longer required in MODE 3. Placing the unit in MODE 3 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 8) assumption that 4 hours is the average time required to perform channel surveillance.

I.1 and I.2

Condition I applies to:

- SG Water Level - High High (P-14) ; and
- Undervoltage Reactor Coolant Pump.

If one channel of SG water level is inoperable, 6 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the Function is then in a partial trip condition where one-out-of-two or one-out-of-three logic will result in actuation. The 6 hour Completion Time is justified in Reference 8. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit to be placed in MODE 3 within the following 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, this Function is no longer required OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, this Function is no longer required OPERABLE.

If one channel of undervoltage reactor coolant pump is inoperable, 6 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the function is then in a

(continued)

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ACTIONS

I.1 and I.2 (continued)

partial trip condition where one additional tripped channel will result in actuation. The 6 hour Completion Time is justified in Ref. 8. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours, requires the Unit to be placed in MODE 2 within the following 6 hours. The allowed Completion time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner without challenging unit systems. In MODE 2, this Function is no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to [4] hours for surveillance testing of other channels. The 6 hours allowed to place the inoperable channel in the tripped condition, and the 4 hours allowed for a second channel to be in the bypassed condition for testing, are justified in Reference 8.

J.1 and J.2 - Not usedK.1, K.2.1 and K.2.2

Condition K applies to RWST Level - Low, which trips both RHR pumps. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to ensure that the Function remains OPERABLE, and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed low). Placing the out-of-service channel in bypass will generate a high level signal on that channel, which will ensure that under no circumstances can a failure of an additional channel low prevent the RHR pumps from starting as the result of an SI signal. The 6 hour Completion Time is justified in Reference 8. If the channel cannot be placed in the bypass condition within 6 hours, and returned to an OPERABLE status within 72 hours, the unit must immediately enter LCO 3.0.3. The 72 hour Allowed Outage Time(AOT) is the same AOT that is allowed for one inoperable RHR pump. This comparison is reasonable because the possible consequences of losing a second level channel can, in the worst case, be no more severe than the loss of one RHR pump, and the probability of losing the level channel is even lower than that of losing an RHR pump. The allowed Completion Times for shutdown are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the unit does not have any analyzed transients or conditions that require the explicit use of the pump trip function noted above.

(continued)

BASES

ACTIONS

K.1, K.2.1 and K.2.2 (continued)

The Required Actions are modified by a Note that allows placing a channel in the bypass condition for up to 4 hours for surveillance testing. The total of 12 hours to reach MODE 3 and 4 hours for a channel to be bypassed is acceptable based on the results of Reference 8.

L.1, L.2.1 and L.2.2

Condition L applies to the P-11 interlock.

With one or more channels inoperable, the operator must verify that the interlock is in the required state for the existing unit condition. This action manually accomplishes the function of the interlock. Determination must be made within 1 hour. The verification determination can be made by observation of the associated annunciator window(s). The 1 hour Completion Time is equal to the time allowed by LCO 3.0.3 to initiate shutdown actions in the event of a complete loss of ESFAS function. If the interlock is not in the required state (or placed in the required state) for the existing unit condition, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of these interlocks.

M.1 or M.2

Condition M applies to the Trip Time Delay (TTD) for the SG low-low water level actuation of AFW pumps. With one or more TTD circuitry delay timers inoperable, 6 hours are allowed to adjust the threshold power level for no time delay to 0% RTP or to place the affected SG water level low-low channel in trip. The specified Completion Time is reasonable considering the nature of these Functions, the available redundancy, and the low probability of an event occurring during this interval. If the TTD threshold power level cannot be adjusted or the affected SG water level low-low channel cannot be placed in trip, the unit must be placed in MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power in an orderly manner and without challenging unit systems. In MODE 4, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above.

(continued)

BASES

ACTIONS
(continued)

N.1 or N.2 and N.2.2

Condition N applies to:

- Manual Initiation of Steam Line Isolation; and
- Manual Initiation of Auxiliary Feedwater.

If a channel is inoperable, 48 hours is allowed to return the channel to an OPERABLE status. The specified Completion Time is reasonable considering the nature of these functions, the available redundancy, and the low probability of an event occurring during this interval. If the Function cannot be returned to OPERABLE status, the associated pump or valve shall be declared inoperable immediately and the REQUIRED ACTION of 3.7.5 or 3.7.2 as applicable complied with immediately.

SURVEILLANCE
REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

Note that each channel of process protection supplies both trains of the ESFAS. When testing channel I, train A and train B must be examined. Similarly, train A and train B must be examined when testing channel II, channel III, and channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(continued)

BASES

ACTIONS

K.1, K.2.1 and K.2.2 (continued)

Agreement criteria are established in STP I-1A, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 31 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.3 - Not usedSR 3.3.2.4

SR 3.3.2.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The time allowed for the testing (4 hours) and the surveillance interval are justified in Reference 8.

SR 3.3.2.5

SR 3.3.2.5 is the performance of a COT.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.2.5 (continued)

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis (Ref. 8) when applicable.

The Frequency of 92 days is justified in Reference 8.

SR 3.3.2.6

SR 3.3.2.6 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function, or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 18 months. The Frequency is adequate, based on operating experience, considering relay reliability and operating history data (Ref. 7)

SR 3.3.2.7 - Not used

SR 3.3.2.8

SR 3.3.2.8 is the performance of a TADOT. This test is a check of the Manual Actuation Functions. It is performed every 18 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.). The Frequency is adequate, based on industry operating experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

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BASES

SURVEILLANCE -
REQUIREMENTS
(continued)

SR 3.3.2.9

SR 3.3.2.9 is the performance of a CHANNEL CALIBRATION.

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.2.10

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the FSAR and SR 3.3.2.10 is only applicable to those functions with a specified limit. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

ESF RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of

(continued)

BASES

SURVEILLANCE -
REQUIREMENTS

SR 3.3.2.10 (continued)

each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 18 months. The 18 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching 650 psig in the SGs.

SR 3.3.2.11

SR 3.3.2.11 is the performance of a TADOT as described in SR 3.3.2.8, except that it is performed for the P-4 Reactor Trip Interlock. The 18 month Frequency is based on operating experience.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

REFERENCES

1. FSAR, Chapter 6.
2. FSAR, Chapter 7.
3. FSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. WCAP-11082, Rev. 2, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Stations - Eagle 21 Version," May 1993
7. WCAP-13900, "Extension of Slave Relay Surveillance Test intervals", April 1994
8. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

(continued)

BASES

REFERENCES
(continued)

9. WCAP-13878, "reliability of Potter & Brumfield MDR Relays",
June 1994.
 10. WCAP-14117, "Reliability Assessment of Potter and Brumfield
MDR Series Relays."
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B 3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified in the FSAR section 7.5 (Ref. 1) based upon the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category I variables.

Type A variables are included in this LCO because they provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for DBAs.

Category I variables are the key variables deemed risk significant because they are needed to:

(continued)

BASES

BACKGROUND
(continued)

- Determine whether other systems important to safety are performing their intended functions;
- Provide information to the operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public, and to estimate the magnitude of any impending threat.

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

APPLICABLE
SAFETY ANALYSES

The PAM instrumentation ensures the operability of Regulatory Guide 1.97 Type A and Category I variables so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures (these variables are restricted to preplanned actions for the primary success path of DBAs), e.g., loss of coolant accident (LOCA);
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- Determine whether systems important to safety are performing their intended functions;
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii). Category I, non-Type A, instrumentation must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category I, non-Type A, variables are important for reducing public risk.

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the unit Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category I, non-Type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident. This capability is consistent with the recommendations of Reference 1.

LCO 3.3.3 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident.

(continued)

BASES

LCO
(continued)

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

The exception to the two channel requirement is Containment Isolation Valve (CIV) Position, Auxiliary Feedwater (AFW) flow indication and Steam Generator (SG) water level (wide range). For the CIV position, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

Both AFW flow indication and SG water level provide indication of the status of decay heat removal capability via the SGs. Sufficient water must be contained in the SGs in order to assure that heat removal can be accomplished through boiling in the SG. Although only one channel exists for each Function, the channels provide diverse indication of the same capability.

Table 3.3.3-1 includes instrumentation which is classified as either Type A and/or Category I variables in accordance with Regulatory Guide 1.97, FSAR Section 7.5, and SER 14.

Type A and Category I variables are required to meet Regulatory Guide 1.97 Category I (Ref. 2) design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display, except as exempted in SSER 31. Regulatory Guide 1.97, for certain Functions, requires that the Function be recorded on at least one channel. For these channels where direct and immediate trend or transient information is not essential for operator information, or both channels would be recorded per Regulatory Guide 1.97, the loss of the recorder is not considered to be a loss of Function. However, the recorder should be returned to service as soon as possible and an alternate means of obtaining the recorded information be established if the recorder is to be out-of service beyond the channel AOT.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.3-1.

(continued)

BASES

LCO
(continued)

1. Neutron Flux (Wide Range NIS)

Neutron Flux indication is provided to verify reactor shutdown. The wide range NIS is necessary to cover the full range of flux that may occur post accident.

Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

2. Steam Line Pressure

Steam pressure is used to determine if a high energy secondary line rupture has occurred and the availability of the steam generators as a heat sink. It is also used to verify that a faulted steam generator is isolated. Steam pressure may be used to ensure proper cooldown rates or to provide a diverse indication for natural circulation cooldown.

3. 4. Reactor Coolant System (RCS) Hot and Cold Leg Temperatures

RCS Hot and Cold Leg Temperatures are Category I variables provided for verification of core cooling and long term surveillance.

RCS hot (outlet) and cold (inlet) leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin will allow termination of safety injection (SI), if still in progress, or reinitiation of SI if it has been stopped. RCS subcooling margin is also used for unit stabilization and cooldown control. RCS hot leg temperature also provides a temperature compensating signal for the reactor vessel level instrumentation system (RVLIS).

In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the unit conditions necessary to establish natural circulation in the RCS. The RCS cold leg temperature also provides a temperature input signal for the low temperature overpressure protection (LTOP) system.

Reactor outlet temperature inputs to the Reactor Protection System are provided by two fast response resistance elements and associated transmitters in each loop. The channels provide indication over a range of 32°F to 700°F.

(continued)

BASES

LCO
(continued)

5. Reactor Coolant System Pressure (Wide Range)

RCS wide range pressure is a Category I variable provided for verification of core cooling and RCS integrity long term surveillance.

RCS pressure is used to verify delivery of SI flow to RCS from at least one train when the RCS pressure is below the pump shutoff head. RCS pressure is also used to verify closure of manually closed spray line valves and pressurizer power operated relief valves (PORVs).

In addition to these verifications, RCS pressure is used for determining RCS subcooling margin. RCS subcooling margin will allow termination of SI, if still in progress, or reinitiation of SI if it has been stopped. RCS pressure can also be used:

- to determine whether to terminate actuated SI or to reinitiate stopped SI;
- to determine when to reset SI and shut off low head SI;
- to manually restart low head SI;
- as reactor coolant pump (RCP) trip criteria; and
- to make a determination on the nature of the accident in progress and where to go next in the procedure.

RCS subcooling margin is also used for unit stabilization and cooldown control.

RCS pressure is also related to three decisions about depressurization. They are:

- to determine whether to proceed with primary system depressurization;
- to verify termination of depressurization; and
- to determine whether to close accumulator isolation valves during a controlled cooldown/depressurization.

Two final uses of RCS pressure are to determine whether to operate the pressurizer heaters and as an input to Reactor Vessel Water Level Instrumentation System (RVLIS).

(continued)

BASES

LCO

5. Reactor Coolant System Pressure (Wide Range) (continued)

RCS pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture (SGTR) or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication. Furthermore, RCS pressure is one factor that may be used in decisions to terminate RCP operation.

6. Reactor Vessel Water Level

RVLIS is provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy.

The RVLIS provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory.

7. a. Containment Wide Range Sump Water Level and
b. Containment Recirculation Sump Level (Narrow Range)

Containment Wide Range Sump Water Level is provided for verification and long term surveillance of RCS integrity.

Containment Recirculation Sump Level (Narrow Range) is used to verify that sufficient water is contained in the recirculation sump to allow operation of the Residual Heat Removal Pumps with the suction aligned to the containment recirculation sump. The required Regulatory Guide 1.97 recorder for this function is part of this instrument channel.

The Wide Range Sump level instrumentation encompasses the range of the Containment Recirculation Sump and can be used to determine the appropriate time for swap-over of the RHR pumps from RWST to the Containment Recirculation Sump if required.

(continued)

BASES

LCO
(continued)

8. a. Containment Pressure (Wide Range) and
b. Containment Pressure (Normal Range)

Containment Pressure is provided for verification of RCS and containment OPERABILITY.

Containment pressure is used to verify closure of main steam isolation valves (MSIVs) during a main steam line break inside containment, and containment spray Phase B isolation when high-high containment pressure is reached.

Both instruments are required to cover the Regulatory Guide 1.97 range requirements.

9. Containment Isolation Valve Position

CIV Position is provided for verification of Containment OPERABILITY, and Phase A and Phase B isolation, and containment ventilation system isolation.

When used to verify Phase A and Phase B isolation, the important information is the isolation status of the containment penetrations. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active CIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve, as applicable, and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE. This Function is on a per valve basis and ACTION A. Is entered separately for each inoperable valve indication. Note (a) to the Required Channels states that the Function is not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(continued)

BASES

LCO
(continued)

10. Containment Area Radiation (High Range)

Containment Area Radiation is provided to monitor for the potential of significant radiation releases for use by operators in determining the need to invoke site emergency plans. Containment radiation level is used to determine if a high energy line break (HELB) containing radioactive fluid has occurred, and whether the event is inside or outside of containment.

11. Containment Hydrogen Concentration

Containment Hydrogen Concentration monitoring is provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions, and is used to determine whether or not hydrogen recombiners should be started.

12. Pressurizer Level

Pressurizer Level is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition.

13. a. Steam Generator Water Level (Wide Range) and
b. Steam Generator Level (Narrow Range)

SG Water Level is provided to monitor operation of decay heat removal via the SGs. The wide range level covers a span of ≥ 6 inches to ≤ 582 inches above the lower tubesheet. The measured differential pressure is displayed in inches of water at 68°F.

SG Water Level is used to:

- identify the faulted SG following a tube rupture;
- verify that the intact SGs are an adequate heat sink for the reactor;
- determine the nature of the accident in progress (e.g., verify an SGTR); and

(continued)

BASES

LCO
(continued)

- verify unit conditions for termination of SI during secondary unit HELBs outside containment.

Operator action is based on the control room indication of SG level. The RCS response during a design basis small break LOCA depends on the break size. For a certain range of break sizes, the reflux cooling mode of heat transfer is necessary to remove decay heat. Wide range level is a Type A variable because the operator must manually raise and control SG level to establish reflux cooling heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated wide range level reaches the reflux cooling initiation point.

SG Water Level (Narrow Range) is redundant to the SG wide range level, and provides indication of adequate RCS heat removal capability during normal SG inventory conditions. The narrow range level covers a span from \approx 437 inches to 581 inches above the lower tubesheet.

14. Condensate Storage Tank (CST) Level

CST Level is provided to ensure water supply for auxiliary feedwater (AFW). The CST provides the ensured safety grade water supply for the AFW System. CST Level is displayed on a control room indicator, strip chart recorder, and unit computer.

CST Level is considered a Type A variable because the control room meter is the primary indication used by the operator.

The DBAs that require AFW are the loss of electric power, steam line break (SLB), and small break LOCA.

The CST is the initial source of water for the AFW System. However, as the CST is depleted, manual operator action is necessary to replenish the CST or align suction to the AFW pumps from the Fire Water Storage Tank or other alternate sources.

15, 16, 17, 18. In-Core Thermocouples

In-Core Thermocouples are provided for verification and long term surveillance of core cooling.

An evaluation was made of the minimum number of valid in-core thermocouples necessary for measuring core cooling.

(continued)

BASES

LCO

15, 16, 17, 18. Core Exit Temperature (continued)

The evaluation determined the reduced complement of in-core thermocouple necessary to detect initial core recovery and trend the ensuing core heatup. The evaluations account for core nonuniformities, including incore effects of the radial decay power distribution, excore effects of condensate runback in the hot legs, and nonuniform inlet temperatures. Based on these evaluations, core cooling can be adequately monitored with two valid in-core thermocouple channels per quadrant with two in-core thermocouples per required channel. Core Exit Temperature is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.

Two OPERABLE channels of In-Core Thermocouples are required in each quadrant to ensure a single failure will not disable the ability to determine the radial temperature gradient.

19. Auxiliary Feedwater (AFW) Flow

AFW Flow is provided to monitor operation of decay heat removal via the SGs.

The AFW Flow to each SG is determined from a differential pressure measurement calibrated for a range of 0 gpm to 300 gpm. Each differential pressure transmitter provides an input to a control room indicator and the unit computer. Since the primary indication used by the operator during an accident is the control room indicator, the PAM specification deals specifically with this portion of the instrument channel.

AFW flow is used three ways:

- to verify delivery of AFW flow to the SGs;
- to determine whether to terminate SI if still in progress, in conjunction with SG water level (narrow range); and
- to regulate AFW flow so that the SG tubes remain covered.

(continued)

BASES

LCO

19. Auxiliary Feedwater Flow (continued)

AFW flow is a Type A variable because operator action is required to throttle flow during an SLB accident to prevent the AFW pumps from operating in runout conditions. AFW flow is also used by the operator to verify that the AFW System is delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level (Narrow Range) during normal SG inventory conditions.

20. Refueling Water Storage Tank (RWST) Water Level

RWST water level is used to verify the water source availability to the emergency core cooling system (ECCS) and Containment Spray Systems. It may also provide an indication of time for initiating cold leg recirculation from the sump following a LOCA. The RWST level signal trips the Residual Heat Removal Pumps at 33% in preparation for transfer to cold leg recirculation.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3 except for the Containment hydrogen Concentration monitor that is only required to be OPERABLE in MODES 1 and 2. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, and in MODE 2 for the Containment Hydrogen Concentration monitor, unit conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

Note 1 has been added in the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident using alternate instruments and methods, and the low probability of an event requiring these instruments.

Note 2 has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

ACTIONS
(continued)

A.1

Condition A applies when one or more Functions have one required channel that is inoperable, but at least one OPERABLE remaining channel. Required Action A.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

B.1

Condition B applies when the Required Action and associated Completion Time for Condition A are not met. This Required Action specifies initiation of actions in Specification 5.6.8, which requires a written report to be submitted to the NRC immediately. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of unit conditions that would require information provided by this instrumentation.

C.1

Condition C applies when one or more Functions have no OPERABLE channels. Required Action C.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with no required channels OPERABLE in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration

(continued)

BASES

ACTIONS

C.1 (continued)

of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur. Condition C is modified by a Note that excludes hydrogen monitor channels.

D.1

Condition D applies when two hydrogen monitor channels are inoperable. Required Action D.1 requires restoring one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable based on the backup capability of the Post Accident Sampling System to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA (which would cause core damage) would occur during this time. Condition D is modified by a Note that limits the APPLICABILITY for the Containment Hydrogen Concentration monitor to MODES 1 and 2.

E.1

Condition E applies when the Required Action and associated Completion Time of Condition C or D are not met. Required Action E.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C or D, and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1 and F.2

If the Required Action and associated Completion Time of Conditions C or D are not met and Table 3.3.3-1 directs entry into Condition F, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

from full power conditions in an orderly manner and without challenging unit systems.

G.1

Alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation will be developed and demonstrated prior to use. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the unit but rather to follow the directions of Specification 5.6.8, in the Administrative Controls section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

H.1

If the Required Action and associated Completion Time of Conditions D is not met and Table 3.3.3-1 directs entry into Condition H, the unit must be brought to a MODE where the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect

(continued)

BASES

SURVEILLANCE -
REQUIREMENTS

SR 3.3.3.1 (continued)

gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized. The Containment Hydrogen Concentration monitors are maintained in a "standby" condition which does not energize all of the monitor components, thus the monitors are not considered "normally energized".

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter with the necessary range and accuracy. This SR is modified by two Notes. Note 1 excludes neutron detectors from CHANNEL CALIBRATION. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." Note 2 discusses an allowed methodology for calibrating the Containment Radiation Level (High Range) Function. The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. FSAR, 7.5.
2. Regulatory Guide 1.97, Revision 3.
3. NUREG-0737, Supplement 1, "TMI Action Items."

(continued)

BASES

REFERENCES
(continued)

4. Supplemental Safety Evaluation Report 14.
 5. Supplemental Safety Evaluation Report 31.
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B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator (SG) safety valves can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System allows extended operation in MODE 3 until such time that either control is transferred back to the Control Room or a cooldown is initiated from outside the control room.

If the control room becomes inaccessible, the operators can establish control at the remote shutdown panel (hot shutdown panel) and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located at the hot shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the remote shutdown control and instrumentation functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible.

APPLICABLE SAFETY ANALYSES

The Remote Shutdown Instrumentation Functions and the hot shutdown panel controls provides equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 3.

The criteria governing the design and specific system requirements of the Remote Shutdown Instrumentation Functions and controls are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The Remote Shutdown Instrumentation Functions and the hot shutdown panel controls is considered an important contributor to the reduction of unit risk to accidents and as such it has been retained in the Technical Specifications as indicated by Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The Remote Shutdown Instrumentation Functions and the hot shutdown panel controls LCO provides the OPERABILITY requirements of the instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation and controls required are listed in Table 3.3.4-1 in the accompanying LCO.

The controls, instrumentation, and transfer switches are required for:

- Reactor trip indication;
- RCS pressure control;
- Decay heat removal via the AFW System and the SG safety valves;
- RCS inventory control via charging flow; and
- Safety support systems for the above Functions, including auxiliary saltwater, component cooling water, and diesel generators.

A Function of a Remote Shutdown System is OPERABLE if all required instrument and control channels for that function listed in Table 3.3.4-1 are OPERABLE. In some cases, Table 3.3.4-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Function is OPERABLE as long

(continued)

BASES

LCO
(continued) as one channel of any of the alternate information or control sources is OPERABLE.

The remote shutdown instrument and control circuits covered by this LCO do not need to be energized to be considered OPERABLE. This LCO is intended to ensure the instruments and control circuits will be OPERABLE if unit conditions require that the Remote Shutdown System be placed in operation.

APPLICABILITY

The Remote Shutdown System LCO is applicable in MODES 1, 2, and 3. This is required so that the unit can be placed and maintained in MODE 3 for an extended period of time from a location other than the control room until either control is transferred back to the control room or a cooldown is initiated.

This LCO is not applicable in MODE 4, 5, or 6. In these MODES, the facility is already subcritical and in a condition of reduced RCS energy. Under these conditions, considerable time is available to restore necessary instrument control functions if control room instruments or controls become unavailable.

ACTIONS

Note 1 is included which excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the low probability of an event requiring the Remote Shutdown System and because the equipment can generally be repaired during operation without significant risk of spurious trip.

Note 2 has been added to the ACTIONS to clarify the application of Completion Time rules. Separate Condition entry is allowed for each Function listed on Table 3.3.4-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

(continued)

BASES

ACTIONS
(continued)

A.1

Condition A addresses the situation where one or more required Functions of the Remote Shutdown Instrumentation and SD panel controls are inoperable. This includes any Function listed in Table 3.3.4-1, as well as the control and transfer switches.

The Required Action is to restore the required Function to OPERABLE status within 30 days. The Completion Time is based on operating experience and the low probability of an event that would require evacuation of the control room.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full-power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.3.4.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

A channel check of the RTBs is inappropriate since their indication is local and any gross failure would be readily detected.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If the channels are

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.4.1 (continued)

within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when Surveillance is required, the CHANNEL CHECK will verify only that they are off scale in the same direction. Offscale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency of 31 days is based upon operating experience which demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.4.2

SR 3.3.4.2 verifies each required Remote Shutdown System control circuit and transfer switch performs the intended function. This verification is performed from the hot shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. (However, this Surveillance is not required to be performed only during a unit outage.) Operating experience demonstrates that remote shutdown control channels usually pass the Surveillance test when performed at the 18 month Frequency.

SR 3.3.4.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The channel calibration is not applicable to the RTB indication.

The Frequency of 18 months is based upon operating experience and consistency with the typical industry refueling cycle.

NOTE: A surveillance of the reactor trip breaker OPERABILITY is not required as part of the SURVEILLANCE REQUIREMENT since a TRIP ACTUATING DEVICE OPERATIONAL TEST of the reactor trip breakers is performed as part of the SURVEILLANCE REQUIREMENT for TS 3.3.1.

(continued)

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.
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B 3.3 INSTRUMENTATION

B 3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or is degraded below a point that would allow safe unit operation. Undervoltage protection will generate an LOP start if a loss of voltage or degraded voltage condition occurs on the 4.16kV vital bus. There are three LOP start signals, one for each 4.16 kV vital bus.

Three undervoltage relays are provided on each 4160 Class 1E vital bus for detecting sustained degraded voltage condition or a loss of bus voltage. A relay will generate an LOP signal (first level undervoltage type relay setpoint) if the voltage is below 75% for a short time. The DG start relays (one per bus) have an inverse time characteristic and will generate an LOP signal with a ≤ 0.8 sec time delay at ≥ 0 volts and at ≤ 10 seconds for ≥ 2583 volts. In addition, the circuit breakers for all loads, except the 4160-480 V load center transformers, are opened automatically by a similar set of first level undervoltage relays. Each of the vital 4160 kV buses has a separate pair of these relays. The relays have a two-out-of-two logic arrangement for each bus to prevent inadvertent tripping of operating loads during a loss of voltage either from a single failure in the potential circuits or from human error. One relay trips instantaneously at ≥ 2870 volts. The second of the two relays has an inverse time characteristic and a delay of ≤ 4 seconds at no voltage and a ≤ 25 second delay with ≥ 2583 volts to prevent loss of operating loads during transient voltage dips, and to permit the offsite power sources to pick up the load. The LOP start actuation is described in FSAR, Section 8.3 (Ref. 1).

Should there be a degraded voltage condition, where the voltage of the vital 4160 kV buses remains at approximately 3785 kV or below, but above the setpoints of the first level undervoltage relays, the following second level undervoltage actions occur automatically:

- (1) After a ≤ 10 second time delay, the respective diesel generators will start.
- (2) After a ≤ 20 second time delay, if the undervoltage condition persists, the circuit breakers for all loads to the respective vital 4160 kV buses, except the 4160-480 V load center transformer, are opened and sequentially loaded on the DG.

Each vital 4160 kV bus has two second level undervoltage relays and one associated timer to initiate each of the above actions (1) and (2) (one timer for each action).

(continued)

BASES

BACKGROUND
(continued)

Trip Setpoints and Allowable Values

The Trip Setpoints used in the relays are based on the analytical limits presented in FSAR, Chapter 15 (Ref. 2). The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account.

The actual nominal Trip Setpoint entered into the relays is normally still more conservative than that required by the Allowable Value. If the measured setpoint does not exceed the Allowable Value, the relay is considered OPERABLE.

Setpoints adjusted in accordance with the Allowable Value ensure that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the accident and that the equipment functions as designed.

Allowable Values and/or Trip Setpoints are specified for each Function in the LCO. Nominal Trip Setpoints are also specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by the surveillance procedure does not exceed the Allowable Value if the relay is performing as required. If the measured setpoint does not exceed the Allowable Value, the relay is considered OPERABLE. Operation with a Trip Setpoint less conservative than the nominal Trip Setpoint, but within the Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculation. Each Allowable Value and/or Trip Setpoint specified is more conservative than the analytical limit assumed in the transient and accident analyses in order to account for instrument uncertainties appropriate to the trip function. These uncertainties are defined WCAP-11082, Rev. 2 "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Stations-Eagle 21 Version " (Ref. 3).

APPLICABLE
SAFETY ANALYSES

The LOP DG start instrumentation is required for the Engineered Safety Features (ESF) Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS).

Accident analyses credit the loading of the DG based on the loss of offsite power during a loss of coolant accident (LOCA). The actual DG start has historically been associated with the ESFAS actuation. The DG loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. The analyses assume a non-mechanistic DG loading, which does not explicitly

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

account for each individual component of loss of power detection and subsequent actions.

The required channels of LOP DG start instrumentation, in conjunction with the ESF systems powered from the DGs, provide unit protection in the event of any of the analyzed accidents discussed in Reference 2, in which a loss of offsite power is assumed.

The delay times assumed in the safety analysis for the ESF equipment include the 10 second DG start delay, and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," include the appropriate DG loading and sequencing delay.

The LOP DG start instrumentation channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO for LOP DG start instrumentation requires that one channel per bus of loss of voltage and two channels per bus for initiation of load shed and two channels per bus of degraded voltage with one timer per bus for DG start and initiation of load shed functions shall be OPERABLE in MODES 1, 2, 3, and 4 when the LOP DG start instrumentation supports safety systems associated with the ESFAS. In MODES 5 and 6, the channels must be OPERABLE whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed. Loss of the LOP DG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power the DG powers the motor driven auxiliary feedwater pumps. Failure of these pumps to start would leave only one turbine driven pump, as well as an increased potential for a loss of decay heat removal through the secondary system.

APPLICABILITY

The LOP DG Start Instrumentation Functions are required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE so that it can perform its function on an LOP or degraded power to the vital bus.

ACTIONS

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the channel is found

(continued)

BASES

ACTIONS
(continued)

inoperable, then the function that channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected.

Because the required channels are specified on a per bus basis, the Condition may be entered separately for each bus as appropriate.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in the LCO. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies when one or more of the loss of voltage or the degraded voltage channel functions (this includes both relays and timers) on a single bus are inoperable.

In these circumstances the Conditions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, "AC Sources - Shutdown," for the DG made inoperable by failure of the LOP instrumentation are required to be entered immediately. The actions of those LCOs provide for adequate compensatory actions to assure unit safety.

A Note is added to allow bypassing an inoperable channel for up to 2 hours for surveillance testing. This allowance is made where bypassing the channel does not cause an actuation and where at least one other channel is monitoring that parameter.

SURVEILLANCE
REQUIREMENTS

SR 3.3.5.2

SR 3.3.5.2 is the performance of a TADOT. This test is performed every 18 months. The test checks trip devices that provide actuation signals directly, bypassing the analog process control equipment. For these tests, the relay Trip Setpoints are verified and adjusted as necessary. The Frequency is based on the known reliability of the relays and controls and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.5.3

SR 3.3.5.3 is the performance of a CHANNEL CALIBRATION.

The setpoints, as well as the response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay.

A CHANNEL CALIBRATION is performed every 18 months. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of 18 months is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. FSAR, Section 8.3.
 2. FSAR, Chapter 15.
 3. WCAP-11082, Rev. 2. "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Stations-Eagle 21 Version". May 1993.
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B 3.3 INSTRUMENTATION

B 3.3.6 Containment Purge and Exhaust Ventilation Isolation Instrumentation

BASES

BACKGROUND

Containment purge and exhaust ventilation isolation instrumentation closes the containment ventilation isolation valves. This action in conjunction with a Phase A signal isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Purge or Vacuum/Pressure Relief System may be in use during reactor operation or reactor shutdown.

Containment purge and exhaust ventilation isolation initiates on a automatic safety injection (SI) signal through the Containment Isolation-Phase A Function, or by manual actuation of Phase A Isolation. The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these modes of initiation.

Two radiation monitoring channels are also provided as input to the containment purge and exhaust ventilation isolation. The two channels measure containment radiation in the exhaust duct for fan E-3. Both detectors will respond to events that release radiation to containment. Both monitors are gaseous activity monitors that will respond to noble gases, particulate and Iodine. The high alarm setpoint is based upon the design basis fuel handling accident source term which does not have a particulate component. The actual high alarm setpoint is more than a factor of 500 below the design calculation earliest actuation point. Since the monitors can only be adjusted to one high alarm setpoint and no particulate is expected during a fuel handling accident, a setpoint based on site boundary noble gases is conservative. For the purposes of this LCO the channels are redundant.

A high radiation signal from either of the two channels initiates containment ventilation isolation, which closes the containment ventilation isolation valves. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event, within approximately 60 seconds. The isolation of the containment ventilation valves has not been analyzed mechanistically in the dose calculations, although its isolation, using a conservative isolation time, is assumed. The containment purge and exhaust ventilation isolation radiation monitors act as backup to the SI signal to ensure closing of the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

purge and exhaust containment ventilation isolation valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The containment purge and exhaust ventilation isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Purge and Exhaust Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation - Not used
2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI, and ESFAS Function 3.a, Containment Phase A Isolation. The applicable MODES and specified conditions for the containment ventilation isolation portion of these Functions are different and less restrictive than those for their Phase A isolation and SI roles. If one or more of the SI or Phase A isolation Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to their SI and Phase A isolation Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Ventilation Isolation remains OPERABLE in MODES 1-4. The LCO requires only one monitor to be OPERABLE during CORE ALTERATIONS or during movement of irradiated fuel.

(continued)

BASES

LCO
(continued)

4. Containment Isolation - Phase A

Refer to LCO 3.3.2, Function 3.a., for all initiating Functions and requirements.

APPLICABILITY

The Automatic Actuation Logic and Actuation Relays, Containment Isolation - Phase A, and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4, and during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment purge and exhaust ventilation isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, the containment purge and exhaust ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment ventilation isolation radiation monitor channel.

(continued)

BASES

ACTIONS

A.1 (continued)

The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that the remaining channel will respond.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

B.1

Condition B applies to all Containment Purge and Exhaust Ventilation Isolation Functions and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for these Functions. It also addresses the failure of both radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, both radiation channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

C.1 and C.2

Condition C applies to all Containment Purge and Exhaust Ventilation Isolation Functions and addresses the train orientation of the SSPS and the master and slave relays for these Functions. It also addresses the condition of no OPERABLE radiation monitoring channels. If a train is inoperable, or the required radiation monitor is inoperable, operation may continue as long as the Required Action to place and maintain containment purge and exhaust ventilation isolation valves (RCV-11, 12, FCV 660, 661, 662, 663, 664) in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Purge and Exhaust Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.6.2 (continued)

and there is an intact voltage signal path to the master relay coils. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

SR 3.3.6.3

SR 3.3.6.3 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

SR 3.3.6.4

A CFT is performed every 92 days on each required channel to ensure the entire channel will perform the intended function. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). This test verifies the capability of the instrumentation to provide the containment purge and exhaust system isolation.

SR 3.3.6.5

SR 3.3.6.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 92 days. The Frequency is acceptable based on instrument reliability and industry operating experience.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.6.6

There is no manual actuation of CVI except via phase A or B. This testing is performed as part of 3.3.2.

SR 3.3.6.7

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

SR 3.3.6.8

This SR assures that the individual channel RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the FSAR. Individual component response times are not modeled in the analyses. The analyses model the overall or elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., valves in full closed position). The response time may be measured by a series of overlapping tests such that the entire response time is measured.

RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every 18 months. The 18 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. 10 CFR 100.11.
 2. NUREG-1366, December 1992.
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B 3.3 INSTRUMENTATION

B 3.3.7 Control Room Ventilation System (CRVS) Actuation Instrumentation

BASES

BACKGROUND

The CRVS provides an enclosed control room environment from which both units can be operated following an uncontrolled release of radioactivity. Upon receipt of an actuation signal, the CRVS shifts from normal operation and initiates filtered ventilation and pressurization of the control room. This system is described in the Bases for LCO 3.7.10, "Control Room Ventilation System", and is common to both units.

The actuation instrumentation consists of redundant radiation monitors in the air intakes to the control room areas. There are two detectors in each of the two normal control room air intakes. However, since they take suction from a common area, the North and South sides of the mechanical equipment room, only two detectors are required to provide protection against a single failure. A Phase "A" containment isolation signal or a high radiation signal from either of the required detectors in the normal intake will initiate CRVS pressurization from the pressurization intake with the lowest radiation level (each pressurization intake, one on the North end of the turbine building and one on the South, has two radiation monitors). The control room operator can also initiate CRVS pressurization by manual switches in the control room.

The CRVS has two additional manually selected operating modes: smoke removal and recirculation. Neither of modes are required for the CRVS to be OPERABLE, but they are useful for certain non-DBA circumstances.

APPLICABLE
SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The CRVS acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

In MODES 1, 2, 3, and 4, the radiation monitor actuation of the CRVS is a backup for the Phase A signal actuation. This ensures initiation of the CRVS during a loss of coolant an accident or steam generator tube rupture involving a release of radioactive materials.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The radiation monitor actuation of the CREFS in MODES 5 and 6, during movement of irradiated fuel assemblies, and CORE ALTERATIONS, is the primary means to ensure control room habitability in the event of a fuel handling or waste gas decay tank rupture accident. The CRVS pressurization system actuation instrumentation satisfies Criterion 310 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CRVS pressurization system is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the CRVS pressurization mode at any time by using two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Actuation Logic and Relays OPERABLE to ensure that no single random failure can prevent automatic actuation of the pressurization system.

3. Control Room Radiation

The LCO specifies two required Control Room Normal Intake Radiation Monitors to ensure that the radiation monitoring instrumentation necessary to initiate the CRVS pressurization system remains OPERABLE.

APPLICABILITY

The CRVS Functions must be OPERABLE in MODES 1, 2, 3, 4, and during CORE ALTERATIONS and movement of irradiated fuel assemblies. The Functions must also be OPERABLE in MODES 5 and 6 when required for a waste gas decay tank rupture accident, to ensure a habitable environment for the control room operators.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient

(continued)

BASES

ACTIONS
(continued)

to exceed the tolerance allowed by the unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CFT and/or Channel Calibration, when the process instrumentation is set up for adjustment to bring it within specification. Drift can also be observed during a Channel check or CFT and if observed would prompt action to correct the discrepancy. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.7-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the actuation logic train Function of the CRVS, the radiation monitor channel Functions, and the manual channel Functions.

If one train is inoperable, or one radiation monitor channel is inoperable in one or more Functions, 7 days are permitted to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this Completion Time is the same as provided in LCO 3.7.10. If the channel/train cannot be restored to OPERABLE status, one CRVS train must be placed in the pressurization mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

(continued)

BASES

ACTIONS
(continued)

B.1.1, B.1.2, and B.2

Condition B applies to the failure of two CRVS actuation trains, two radiation monitor channels, or two manual channels. The first Required Action is to place one CRVS train in the pressurization mode of operation immediately. This accomplishes the actuation instrumentation Function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.10 must also be entered for the CRVS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.10.

Alternatively, both trains may be placed in the pressurization mode. This ensures the CREFS function is performed even in the presence of a single failure.

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS
(continued)

D.1 and D.2

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met during CORE ALTERATIONS or when irradiated fuel assemblies are being moved. Movement of irradiated fuel assemblies and CORE ALTERATIONS must be suspended immediately to reduce the risk of accidents that would require CREFS actuation.

E.1

Condition E applies when the Required Action and associated Completion Time for Condition A or B have not been met in MODE 5 or 6. Actions must be initiated to restore the inoperable train(s) to OPERABLE status immediately to ensure adequate isolation capability in the event of a waste gas decay tank rupture.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CREFS Actuation Functions.

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.7.1 (continued)

including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.7.2

A CFT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the CRVS actuation. The Frequency is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.7.3

SR 3.3.7.3 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 31 days on a STAGGERED TEST BASIS. The Frequency is justified in WCAP-10271-P-A, Supplement 2, Rev. 1 (Ref. 1).

SR 3.3.7.4

SR 3.3.7.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.7.4 (continued)

check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The Frequency is acceptable based on instrument reliability and industry operating experience.

SR 3.3.7.5

SR 3.3.7.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation MODE is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation MODE is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 18 months. The Frequency is acceptable based on instrument reliability and operating experience (Ref. 1 and 2).

SR 3.3.7.6

SR 3.3.7.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every 18 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The test also includes trip devices that provide actuation signals directly to the Solid State Protection System, bypassing the analog process control equipment. The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience. The SR is modified by a Note that excludes verification of setpoints during the

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.3.7.6 (continued)

TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.7.7

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. WCAP-13878. "reliability of Potter & Brumfield MDR Relays", June 1994.
 2. WCAP-13900. "Extension of Slave Relay Surveillance Test intervals", April 1994
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B 3.3 INSTRUMENTATION

B 3.3.8 Fuel Handling Building Ventilation System (FHBVS) Actuation Instrumentation

BASES

BACKGROUND

The FHBVS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.13, "Fuel Handling Building Ventilation System." The system initiates filtered ventilation of the fuel building automatically following receipt of a high radiation signal from the Spent Fuel Pool Monitor or from the New Fuel Storage Vault Monitor (or from gaseous monitors 45 A/B when installed). Initiation may also be performed manually as needed from the main control room or fuel handling building.

High radiation, from either of the two monitors, provides FHBVS initiation. radiation detected by any monitor. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the fuel building.

APPLICABLE
SAFETY ANALYSES

The FHBVS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the fuel building exhaust following a fuel handling accident so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1).

The FHBVS actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO The LCO requirements ensure that instrumentation necessary to initiate the FHBVS is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate the FHBVS at any time by using either of two switches, one in the control room and another in the fuel handling building. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

2. Fuel Building Radiation

The LCO specifies two required Radiation Monitor channels to ensure that the radiation monitoring instrumentation necessary to initiate the FBVS remains OPERABLE.

Only the Trip Setpoint is specified for each FHBVS Function in the LCO. The Trip Setpoint limits account for instrument uncertainties, which are defined in the Unit Specific Setpoint Calibration Procedure (Ref. 2).

APPLICABILITY

The manual FBVS initiation must be OPERABLE when moving irradiated fuel assemblies in the fuel building, to ensure the FHBVS operates to remove fission products associated with a fuel handling accident.

High radiation initiation of the FHBVS must be OPERABLE in any MODE during movement of irradiated fuel assemblies in the fuel building to ensure automatic initiation of the FBVS when the potential for a fuel handling accident exists.

While in MODES 5 and 6 without fuel handling in progress, the FHBVS instrumentation need not be OPERABLE since a fuel handling accident cannot occur.

(continued)

BASES (continued)

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a CFT and/or Channel calibration, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered. Drift can also be observed during a Channel check or CFT and if observed would prompt action to correct the discrepancy.

Two Notes have been added to the ACTIONS to clarify the application of Completion Time rules and the Applicable of LCO 3.0.3. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in Mode 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1

Condition A applies to the radiation monitor functions, and the manual function. Condition A applies to the failure of one or more radiation monitor channels, or manual channels. If one or more channels or trains are inoperable, a period of 30 days is allowed to restore it to OPERABLE status. If the train cannot be restored to OPERABLE status, an appropriate portable continuous monitor with the same setpoint, or an individual qualified in radiation protection procedures with a dose rate monitoring device must be in the spent fuel pool area, one FHBVS train must be placed in in the Iodine Removal mode of operation immediately. This effectively accomplishes the actuation instrumentation function and places the area in a conservative mode of operation or provides appropriate monitoring for continued fuel movement.

(continued)

BASES

ACTIONS
(continued)

C.1

Condition C applies when the Required Action and associated Completion Time for Condition A has not been met and irradiated fuel assemblies are being moved in the fuel building. Movement of irradiated fuel assemblies in the fuel building must be suspended immediately to eliminate the potential for events that could require FBVS actuation.

SURVEILLANCE
REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.8-1 determines which SRs apply to which FHBVS Actuation Functions.

Notes have been added that clarify which functions will be associated with which monitors when the new radiation monitors RM-45A and 45B are installed.

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.3.8.2

A CFT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the FBACS actuation. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.8.3 - Not used

SR 3.3.8.4

SR 3.3.8.4 is the performance of a TADOT. This test is a check of the manual actuation functions and is performed every 18 months. Each manual actuation function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Frequency is based on operating experience and is consistent with the typical industry refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.8.5

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling
(DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses and is variable with reactor thermal power down to 90% RTP as shown on Tables 3.4.1-1 and 3.4.1-2. Flow rate indications from the plant computer or RCS flow rate indicators are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the DNB limits to be approached.

Operation for significant periods of time outside the limits on RCS flow, pressurizer pressure and average RCS temperature increases the likelihood of a fuel cladding failure if a DNB limited event were to occur.

APPLICABLE
SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR correlation limit of ≥ 1.17 (Ref. 2 and 3). This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

impact on the DNBR criterion. The analyzed transients include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)", and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of 2197.3 psig and the RCS average temperature limit of 584.3°F correspond to nominal analytical limits of 2250 psia and 577.6°F for Unit 2 (the limiting unit) used for the DNB calculation in the reload analyses with allowance for analysis initial consideration uncertainty (38 psi and 6.7°F).

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

This LCO specifies limits on the monitored process variables--pressurizer pressure, RCS average temperature, and RCS total flow rate to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow limits are provided for a RTP range of 90% to 100% on Tables 3.4.1-1 and 3.4.1-2 for Unit 1 and Unit 2 respectively.

The RCS total flow rate limit allows for a measurement error of 2.34% based on performing a precision heat balance and using the result to normalize the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a non-conservative manner. Therefore, a bias error of 0.1% for undetected fouling of the feedwater venturi is included in the measurement error analysis.

Any fouling that might significantly bias the flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The LCO numerical values for pressure, temperature, and flow rate have not been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or

(continued)

BASES

other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational pressure transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

Another set of limits on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and reduce the potential for violation of the accident analysis limits.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition reduces the potential for violation of the accident analysis limits. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

The 12 hour Surveillance Frequency for the indicated RCS total flow rate is performed using the installed flow instrumentation. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation within safety analysis assumptions. The term "indicated RCS total flow" is used to distinguish between the "measured RCS total flow" determined in SR 3.4.1.4.

SR 3.4.1.4

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance or other acceptable method once every 18 months allows the installed RCS flow instrumentation to be normalized and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance. Flow verification demonstrates that setpoints are relevant and RCS flow resistance is within limits.

(continued)

BASES

REFERENCES

1. FSAR, Section 15.
 2. Diablo Canyon Power Plant Unit 1 Cycle 9 Reload Safety Evaluation, August 1995.
 3. Diablo Canyon Power Plant Unit 2 Cycle 8 Reload Safety Evaluation, Rev.1, April 1996.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

APPLICABLE
SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

All low power safety analyses assume initial RCS loop temperatures \geq the HZP temperature of 547°F (Ref. 1). The

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

minimum temperature for criticality limitation provides a small band, 6°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) with an operating loop temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

The special test exception of LCO 3.1.8, "PHYSICS TESTS, Exceptions, MODE 2" permits PHYSICS TESTS to be performed at $\leq 5\%$ RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with $k_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with $k_{eff} < 1.0$ in an orderly manner and without challenging plant systems.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above 541 °F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours is frequent enough to prevent inadvertent violation of the LCO and takes into account indications and alarms that are continuously available to the operator in the control room. If the $T_{avg}-T_{ref}$ deviation were to alarm, the specific alarm response procedure would provide an increased frequency of monitoring. Following the clearance of the alarm, the frequency returns to 12 hours to monitor RCS T_{avg} .

REFERENCES

1. FSAR; Chapter 15.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PRESSURE TEMPERATURE LIMITS REPORT (PTLR) contains pressure/temperature (P/T) limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO references the PTLR which establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the PTLR limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. The NRC reviewed and approved methodology to be applied to determine P/T Limits is documented in the Administrative Controls Section 5.6.6.

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with the methodology identified in Section 5.6.6. The operating P/T limit curves will be adjusted, as necessary, in agreement with the evaluation findings based on methods used in the PTLR.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and

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BASES

BACKGROUND
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temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be $\geq 40^\circ\text{F}$ above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components.

APPLICABLE
SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Administrative Controls Section 5.6.6, identifies the NRC reviewed and approved methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

The two elements of this LCO are:

- a. The limit curves for heatup, cooldown, and ISLH testing; and
- b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and

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BASES

LCO
(continued)

permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

- a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;
- b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and
- c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

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BASES

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 3), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

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BASES

ACTIONS
(continued)

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. (Note that Action B.1 is not required when in MODE 4.) A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 3), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. Not Used
 2. 10 CFR 50, Appendix G.
 3. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E
 4. NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

- a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;
- b. Improving the neutron economy by acting as a reflector;
- c. Carrying the soluble neutron poison, boric acid;
- d. Providing a second barrier against fission product release to the environment; and
- e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The reactor coolant is circulated through four loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor core and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES.

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

All of the accident/safety analyses performed at RTP assume that all four RCS loops are in operation as an initial condition. Some accident/safety analyses have been performed at zero power conditions assuming only two RCS loops are in operation to conservatively bound lower modes of operation. The uncontrolled Rod Control Cluster Assembly (RCCA) Bank withdrawal from

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

subcritical event is included in this category. While all accident/safety analyses performed at full rated power assume that all RCS loops are in operation, selected events examine the effects resulting from a loss of RCP operation. These include the complete and partial loss of forced RCS flow, RCP rotor seizure, and RCP shaft break events. For each of these events, it is demonstrated that all the applicable safety criteria are satisfied. For the remaining accident/safety analyses, operation of all four RCS loops during the transient up to the time of reactor trip is assured thereby ensuring that all the applicable acceptance criteria are satisfied. Those transients analyzed beyond the time of reactor trip were examined assuming that a loss of offsite power occurs which results in the RCPs coasting down.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the Safety Limit value during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

An OPERABLE RCS loop consists of one OPERABLE RCP for heat transport and the associated SG, OPERABLE in accordance with the Steam Generator Tube Surveillance Program, with a water level within the limits specified in SR 3.4.5.2, except for operational transients. A RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink

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BASES

APPLICABILITY
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requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

- LCO 3.4.5, "RCS Loops - MODE 3";
 - LCO 3.4.6, "RCS Loops - MODE 4";
 - LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
 - LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
 - LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
 - LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).
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ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours that each RCS loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

REFERENCES

1. FSAR, Section 15.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor core and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

APPLICABLE SAFETY ANALYSES

Whenever the Control Rod Drive Mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the Rod Control System. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible. Such a transient could be caused by the mechanical failure of a CRDM.

Therefore, in MODE 3 with the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least two RCS loops to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

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BASES

RCS Loops-MODE 3 satisfy Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

The purpose of this LCO is to require that at least two RCS loops be OPERABLE. In MODE 3 with the Rod Control System capable of rod withdrawal, two RCS loops must be in operation. Two RCS loops are required to be in operation in MODE 3 with the Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit (SL) criteria will be met for all of the postulated accidents.

With the CRDMs de-energized, the Rod Control System is not capable of rod withdrawal; therefore, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that redundancy for heat removal is maintained.

The Note permits all RCPs to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are required to be performed without flow or pump noise. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

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BASES

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops-MODES 1 and 2";
LCO 3.4.6, "RCS Loops-MODE 4";
LCO 3.4.7, "RCS Loops-Mode 5, Loops Filled";
LCO 3.4.8, "RCS Loops-Mode 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

B.1

If restoration is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal (RHR) System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If the required RCS loop is not in operation and the Rod Control System is capable of rod withdrawal, the Required Action is either to restore the required RCS loop to operation or to place the Rod Control System in a condition incapable of rod

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BASES

ACTIONS
(continued)

withdrawal, (e.g., de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets.) When the Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the Rod Control System must not be capable of rod withdrawal. The Completion Time of 1 hour to restore the required RCS loop to operation or to defeat the Rod Control System is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

D.1, D.2, and D.3

If four RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, place the Rod Control System in a condition incapable of rod withdrawal, (e.g., all CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets.) All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal.

The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE
REQUIREMENTSSR 3.4.5.1

This SR requires verification every 12 hours that the required loops are in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

narrow range water level is $\geq 15\%$ for required RCS loops. If the SG secondary side narrow range water level is $< 15\%$, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

SR 3.4.5.3

Verification that the required RCPs are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through four RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor core and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. For RHR operation, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, a RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR loop circulates the water through the RCS at a sufficient rate to remove decay heat and to prevent boric acid stratification.

Although NUREG-1431 uses "loop" to define RHR system requirements, past practice is use of "train", consistent with ECCS discussions of train availability and redundancy. Plant procedures are written using "train". The designations of "loop" and "train" are considered synonymous.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS Loops - MODE 4 have been identified in 10 CFR 50.36 (c) (2) (ii) as important contributors to risk reduction.

(continued)

BASES

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be removed from operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise. 1 hour is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that the secondary side water temperature of each SG be $< 50^\circ\text{F}$ above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature \leq the temperature below which LTOP is required as specified in the PTLR (the current limiting temperature for DCP is 270°F .) Note 2 also includes a DCP plant specific alternate condition under which a RCP may be started in MODE 4 and in MODE 5 with the loops filled. Note that RCPs may be "bumped" following a condition of RCS depressurization to establish "loops filled" condition. The Note specifies that a RCP may be started if the pressurizer water level is less than 50%. This option of RCP start with pressurizer water level less than 50% supports plant operational flexibility. The open volume in the pressurizer provides space to sustain reactor coolant thermal swell without incurring a possible excessive pressure transient due to energy additions from the S/G secondary water. The purpose of conditions to allow initial RCP start when none is running is to prevent a possible low temperature RCS overpressure event due to a thermal transient when a RCP is started. The condition of SG/RCS temperature difference limits the available relative energy source and the pressurizer level condition provides an expansion volume to

(continued)

BASES

LCO
(continued)

accommodate possible reactor coolant thermal swell. These conditions are intended to prevent a low temperature overpressure event due to a thermal transient when a RCP is started.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required. A RHR loop is in operation when the pump is operating and providing forced flow through the loop. Because a loop can be operating without being OPERABLE, the LCO requires at least one loop OPERABLE and in operation.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one required RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS loop or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

BASES

ACTIONS
(continued)

If one required RHR loop is OPERABLE and in operation and there are no RCS loops OPERABLE, an inoperable RCS loop or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

If the parameters that are outside the limits cannot be restored, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 ($\leq 200^{\circ}\text{F}$) rather than MODE 4 ($> 200^{\circ}\text{F}$ to $< 350^{\circ}\text{F}$). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

B.1 and B.2

If no loop is OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated.

Boron dilution requires forced RCS circulation from at least one RCP for proper mixing, so that an inadvertent criticality may be prevented. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE
REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one RCS loop or RHR loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is $\geq 15\%$. If the SG secondary side narrow range water level is $< 15\%$, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs are specified as a backup means for redundancy. Even though the SG cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or two SGs with secondary side water levels above 15% to provide an alternate method for decay heat removal via natural circulation.

APPLICABLE
SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops - MODE 5 (Loops Filled) have been identified in 10 CFR 50.36 (c)(2)(ii) as important contributors to risk reduction.

(continued)

BASES

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or two SGs with secondary side water level $\geq 15\%$. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is two SGs with their secondary side water levels $\geq 15\%$. Should the operating RHR loop fail, the SGs could be used to remove the decay heat via natural circulation.

Note 1 permits all RHR pumps to be removed from operation ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise. 1 hour is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be $\leq 50^{\circ}\text{F}$ above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with any RCS cold leg temperature \leq the temperature below which Low Temperature Overpressure Protection (LTOP) is required as specified in the PTLR. Note that RCPs may be "bumped" following a condition of RCS depressurization to establish "loops filled" condition.

Note 3 also includes an OR condition for starting a RCP. This condition is a DCPP plant specific alternate condition under which a RCP may be started in MODE 4 and in MODE 5 with the loops

(continued)

BASES

LCO
(continued)

filled. The Note specifies that a RCP may be started if the pressurizer water level is less than 50%. This option of RCP start with pressurizer water level less than 50% supports plant operational flexibility. The open volume in the pressurizer provides space to sustain reactor coolant thermal swell without incurring a possible excessive pressure transient due to energy additions from the S/G secondary water. The purpose of conditions to allow initial RCP start when none is running is to prevent a possible low temperature RCS overpressure event due to a thermal transient when a RCP is started. The condition of SG/RCS temperature difference limits the available relative energy source and the pressurizer level condition provides an expansion volume to accommodate possible reactor coolant thermal swell. These conditions are intended to prevent a low temperature overpressure event due to a thermal transient when a RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be $\geq 15\%$.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

(continued)

BASES

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side water levels < 15%, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Notes 1 and 4, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent inadvertent criticality during a boron dilution, forced circulation from at least one RCP is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE
REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

Verifying that at least two SGs are OPERABLE by ensuring their secondary side narrow range water levels are $\geq 15\%$ ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.7.3

Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side water level is $\geq 15\%$ in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation. The flow provided by one RHR loop is adequate for heat removal and for boron mixing.

RCS loops in MODE 5 (loops not filled) have been identified in 10 CFR 50.36 (c) (2) (ii) as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one running RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to meet single failure considerations.

Note 1 permits all RHR pumps to be removed from operation for ≤ 1 hour. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained at least 10°F below saturation temperature. The Note prohibits boron dilution or draining operations when RHR forced flow is stopped.

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BASES

LCO
(continued)

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System. The Applicability is modified by a Note stating that while the LCO is not met, entry into MODE 5, Loops Not Filled, from MODE 5, Loops Filled, is not permitted. This Note specifies an exception to LCO 3.0.4 and would prevent draining the RCS, which would eliminate the possibility of SG heat removal, while the RHR function was degraded

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no required RHR loops are OPERABLE or in operation, except during conditions permitted by Note 1, all operations involving a reduction of RCS boron concentration must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. Boron dilution requires forced circulation from at least one RCP for proper mixing so that inadvertent criticality can be prevented. The immediate Completion Time

(continued)

BASES

reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

SURVEILLANCE
REQUIREMENTS

SR 3.4.8.1

This SR requires verification every 12 hours that one loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.8.2

Verification that the required number of pumps are OPERABLE ensures that additional pumps can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pumps. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls and emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system.

Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

APPLICABLE
SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensable gases normally present.

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Safety analyses presented in the FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water volume \leq 1600 cubic feet, which is equivalent to 90% of span ensures that a steam bubble exists. Instrument inaccuracy is not included in this % number. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity \geq 150 kW, capable of being powered from either the offsite power source or the emergency power supply. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. The capability to power the heaters from an emergency power supply using bus cross-tie to an OPERABLE emergency diesel generator, if necessary, provides the means to maintain system pressure control during a loss of normal power. RCS pressure control is necessary to maintain subcooling under conditions of natural circulation flow in the primary system. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and

(continued)

BASES

cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

APPLICABILITY
(continued)

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply, and if necessary, using bus cross-tie to an OPERABLE emergency diesel generator. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1, A.2, A.3, and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. The upper limit of this LCO is below the Pressurizer Water Level-High Trip at 92% of span.

If the pressurizer water level is not within the limit, action must be taken to bring the unit to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3, with rods fully inserted and the Rod Control System not capable of rod withdrawal. Additionally, the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

The allowed Completion Times are reasonable; based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1

If one required group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station powered heaters.

C.1 and C.2

(continued)

BASES

If one required group of pressurizer heaters is inoperable and cannot be restored in the allowed Completion Time of Required

ACTIONS
(continued)

Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is consistent with the safety analyses assumptions of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The Frequency of 18 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

SR 3.4.9.3

This SR demonstrates that the heaters can be manually transferred from the normal to the emergency power supply and energized. The Frequency of 18 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.

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BASES

REFERENCES

1. FSAR, Section 15.
 2. NUREG-0737, November 1980.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr at 2485 psig plus 3% accumulation, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves which is divided equally among the three valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves and an increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, and 4. However, in MODE 4, with one or more RCS cold leg temperatures \leq the temperature below which LTOP is required as specified in the PTLR (the current limiting temperature for DCP is 270°F) and in MODE 5 and MODE 6 with the reactor vessel head on and the reactor vessel head closure bolts not fully de-tensioned, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the $\pm 1\%$ of nominal pressure tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot (which is the current DCP practice) or if valves are set cold, that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

(continued)

BASES

BACKGROUND
(continued)

The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE
SAFETY ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include:

- a. Uncontrolled rod withdrawal from full power;
- b. Feedline break;
- c. Loss of external electrical load;
- d. Loss of normal feedwater;
- e. Loss of all AC power to station auxiliaries; and
- f. Locked Reactor Coolant Pump (RCP) rotor.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events b, c, d, e and f (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

LCO

The three pressurizer safety valves are set to open at the RCS design pressure (2485 psig), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the $\pm 1\%$ of nominal pressure tolerance requirements (Ref. 1) for lifting pressures above 1000 psig.

The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS

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BASES

components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 when any RCS cold leg temperature is \leq the temperature below which LTOP is required as specified in the PTLR, or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head closure bolts fully de-tensioned.

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures \leq the temperature below which LTOP is required as specified in the PTLR within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full

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BASES

ACTIONS

(continued)

power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperature at or below the temperature below which LTOP is required as specified in the PTLR, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE
REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief tests be performed in accordance with ANSI/ASME OM-a-1988 (Ref. 5.). No additional requirements are specified. The surveillance specifies the lift settings to be within $\pm 1\%$ of nominal pressure of 2485 psig.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
 2. FSAR, Chapter 15.
 3. WCAP-7769, Rev. 1, June 1972.
 4. ASME, Boiler and Pressure Vessel Code, Section XI.
 5. Operation and Maintenance Code, 1987 with OM-a-1988 Addenda.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air operated valves that are controlled to open when the pressurizer pressure increases above their actuation setpoint and to close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

DCPP design includes three air operated pressurizer PORVs. Two of these PORVs have been designated as "Class I". These two valves provide the reactor vessel low temperature overpressure protection and they provide the means to depressurize the RCS following a steam generator tube rupture (SGTR.) These functions must be accomplished under accident analyses assumptions such as loss of offsite power. Consequently, a Class I nitrogen backup system to the non-safety related air supply is provided for the two Class I PORVs. The identification of Class I is used to make a distinction between these two PORVs that must provide a safety-related function as opposed to the third remaining PORV that is designated as non-Class I. TS 3.4.12 for LTOP applies to the two Class I PORVs but not to the non-Class I PORV.

The non-Class I PORV is an element of the DCPP design for 100% load rejection without reactor trip. This valve is associated with plant transients as compared to accident mitigation. Although mitigation is not its primary purpose, the valve may be used for those functions also, although not credited for operation.

The three PORVs are the same design. The PORV that is not designated as Class I may be used, when instrument air is available, to control RCS pressure similarly to the Class I PORVs. However, two Class I PORVs satisfy the function, with redundancy, therefore continued operation with the non-Class I PORV unavailable for RCS pressure control is allowed as long as the block valve or PORV can be closed to maintain the RCS pressure boundary. However, the plant capability to sustain a 100% load rejection without reactor trip would be compromised.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The three MOV block valves are the same design. The block valves are used to isolate the PORVs in case of excessive seat leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

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BASES

BACKGROUND
(continued)

The PORVs may be manually cycled and are equipped with circuitry for automatic actuation. No credit is taken for PORV automatic actuation in the FSAR analyses for MODE 1, 2 or 3 transients where PORV operation may have a beneficial effect. Therefore the PORVs may be OPERABLE in either manual operation or the automatic mode. The automatic mode is the preferred configuration, as this provides pressure relieving capability without reliance on operator action.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permits performance of surveillances on the block valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. The PORV block valves are all powered from separate vital busses.

The plant has three PORVs, each having a relief capacity of 210,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer Pressure-High reactor trip setpoint up to and including the design step-load decrease. In addition, the PORVs minimize challenges to the pressurizer safety valves and the two Class I PORVs are used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

APPLICABLE
SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal or auxiliary pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes manual operator actions to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. For the SGTR event, the PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

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BASES

Automatic actuation of the PORVs is not assumed in any design basis accidents during MODES 1, 2, and 3.

Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining the PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. The block valves are available to isolate the flow path through either a failed open PORV or a PORV with excessive seat leakage. Satisfying the LCO helps minimize challenges to fission product barriers. Note, however, that operability of the PORVs (as indicated by the surveillances) only requires that the PORVs be capable of being manually cycled to perform their safety function, and that they need not be capable of automatic actuation since that is not a safety function.

APPLICABILITY

In MODES 1, 2, and 3, the PORVs are required to be OPERABLE to mitigate a SGTR and the block valves are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. PORV OPERABILITY in MODES 1, 2, and 3 will also minimize challenges to the pressurizer safety valves.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. OPERABILITY of the PORVs requires them to be capable of manual operation. Automatic operation is not assumed in accident analyses and therefore is not a required safety function. LCO 3.4.11 is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place and the reactor vessel head closure bolts not fully de-tensioned. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

Note 1 has been added to clarify that all pressurizer PORVs are treated as separate entities, each with separate Completion Times

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BASES

ACTIONS
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(i.e., the Completion Time is on a component basis). The exception for LCO 3.0.4, Note 2, permits MODE changes with inoperable PORVs or block valves as one possible recourse to remaining in the Applicability of LCO 3.4.12.

A.1

PORVs may be inoperable and capable of being manually cycled, (e.g., excessive seat leakage) In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valves is required to be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. No credit is given for automatic PORV operation in Reference 2 analyses for MODE 1, 2, and 3 transients. As such, the PORVs are considered OPERABLE in either manual control or in the automatic mode. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV isolation may be necessary due to seat leakage, instrumentation problems, automatic control problems, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. For these reasons, the block valve may be closed but the ACTION requires power be maintained to the valve. This Condition is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition. Normally, the PORVs should be available for automatic mitigation of overpressure events and should be returned to OPERABLE and automatic actuation status prior to entering startup (MODE 2).

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2, and B.3

If one PORV is inoperable and not capable of being manually cycled, it must be either restored or isolated by closing the associated block valve and removing the power to the associated block valve. The Completion Time of 1 hour is reasonable, based on challenges to the PORVs during this time period, and provides the operator adequate time to correct the situation. If the inoperable PORV cannot be restored to OPERABLE status, it must be isolated within the specified time. Because at least one Class I PORV remains OPERABLE, an additional 72 hours is provided to

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ACTIONS
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restore the inoperable PORV to OPERABLE status if it is Class I. If the valve is the non-Class I PORV, there is no required Completion Time. If the Class I PORV cannot be restored within this additional time, the plant must be brought to at least MODE 3 with Tavg less than 500°F, as required by Condition D.

C.1, C.2, and C.3

If one PORV block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The PORV control switch has three positions: open, close, and auto. Placing the PORV in manual control, if required in ACTION C, is accomplished by positioning the switch out of the auto control mode. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the associated PORV in manual control.

This action is taken to avoid the potential for a stuck open PORV if the valve were to open under automatic control at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. If the inoperable block valve is associated with a Class I PORV, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the Class I PORV block valve is based upon the Completion Time for restoring an inoperable Class I PORV in Condition B, since the PORVs are not capable of mitigating a SGTR when inoperable and not capable of being manually cycled. If the block valve is restored within the Completion Time of 72 hours, the PORV will be transferred to the automatic mode of operation. If the block valve cannot be restored within this additional time, the plant must be brought to at least MODE 3 with Tavg less than 500°F as required by Condition D.

If the Inoperable block valve is associated with the non-Class I PORV, the block valve may be closed and the power removed. The 72 hour Completion Time for closing the block valve is the same applied in Required Action C.2. Using the same time recognizes that some restoration work may be required since the block valve is inoperable. Restoration of the non-class I PORV block valve to OPERABLE status is not required because the non-Class I PORV is not required to be available for safety function although having the valve closed impairs the load rejection design capability. Therefore, once the block valve is closed per

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ACTIONS
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Required Action C.3, the Completion Time requirement of Condition D does not apply.

If the block valve can not be placed in the closed position per Required Action C.3., Condition D applies and the unit must be taken to MODE 3 with Tavg less than 500°F until the block valve is restored or closed.

D.1, D.2, and D.3

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a condition below where the function of the PORVs to mitigate a SGTR event is needed. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and Tavg reduced to < 500°F within 12 hours. Additional action is required to be initiated immediately to continue efforts to restore the inoperable valve(s) to OPERABLE status. This action will ensure expedient measures are taken to reestablish OPERABLE PORVs and block valves while maintaining plant conditions above MODE 4 but less than 500°F. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4, 5, and 6 with the reactor vessel head closure bolts not fully de-tensioned, maintaining Class I PORV OPERABILITY is required by LCO 3.4.12.

E.1, E.2, E.3, E.4, and E.5

If more than one Class I PORV is inoperable and not capable of being manually cycled, it is necessary to immediately initiate action to restore the valves and to, within the Completion Time of 1 hour, isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If one Class I PORV is restored and one Class I PORV remains inoperable, then the plant will be in Condition B with the time clock started at the original declaration of having two Class I PORVs inoperable. If no Class I PORVs are restored within the Completion Time, then the plant must be brought to a condition below where the function of the PORVs to mitigate a SGTR event is not needed. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and Tavg reduced to < 500°F within 12 hours.

Additional action is required to be initiated immediately to continue efforts to restore the inoperable valve(s) to OPERABLE status. This action will ensure expedient measures are taken to

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ACTIONS
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reestablish OPERABLE PORVs and block valves while maintaining plant conditions above MODE 4 but less than 500°F. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In

MODES 4, 5, and 6 with the reactor vessel head closure bolts not fully de-tensioned, maintaining Class I PORV OPERABILITY is required by LCO 3.4.12.

F.1, F.2, F.3, and F.4

If more than one PORV block valve is inoperable, it is necessary to either restore the block valves within the Completion Time of 1 hour, or place the associated PORVs in manual control and restore at least one block valve within 2 hours and restore the remaining block valve within 72 hours. The PORV control switch has three positions; open, close and auto. Placing the PORV in manual control, if required in ACTION F, may be accomplished by positioning the switch out of the auto control mode. The Completion Times are reasonable, based on the small potential for challenges to the system during this time and provide the operator time to correct the situation.

If the Inoperable block valve is associated with the non-Class I PORV, the block valve may be closed and the power removed. The 72 hour Completion Time for closing the block valve is the same time used in Required Action F.3. This Completion Time recognizes that some restoration work may be required since the block valve is inoperable. Restoration of the non-class I PORV block valve to OPERABLE status is not required because the non-Class I PORV is not required to be available for safety function, although having the valve closed impairs the load rejection design capability. Therefore, once the block valve has been closed per Required Action F.4, Completion Time requirements of Condition G does not apply.

If the block valve can not be placed in the closed position per Required Action F.4, Condition G applies and the unit must be taken to MODE 3 with Tavg less than 500°F until the block valve is restored or closed.

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a condition below where the function of the PORVs to mitigate a SGTR event is not needed. To achieve this status, the plant must be brought to at least MODE 3 within

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BASES

ACTIONS
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Tavg reduced to < 500°F within 12 hours. Additional action is required to be initiated immediately to continue efforts to restore the inoperable valve(s) to OPERABLE status. This action will ensure expedient measures are taken to reestablish OPERABLE PORVs and block valves while maintaining plant conditions above MODE 4 but less than 500°F. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4, 5, and 6 with the reactor vessel head closure bolts not fully de-tensioned, maintaining Class I PORV OPERABILITY is required by LCO 3.4.12.

SURVEILLANCE
REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the Frequency of 92 days is the ASME O & M Code, Part 10 (Ref. 3). If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an inoperable Class I PORV that is incapable of being manually cycled, the maximum Completion Time to restore the Class I PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve Frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the Class I PORV to OPERABLE status.

If the block valve for a non-class I PORV has been closed by Required Action C.3 or F.4, it can remain closed without further requirement for restoration. The non-Class I PORV is not required for accident mitigation and the closed block valve provides protection against a spurious opening of the PORV.

Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed in accordance with the Required Action of Condition A., B. and E., or the Required Action C.3 and F.4.

Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

manually actuated for mitigation of an SGTR. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2. In accordance with Reference 4, administrative controls require this test be performed in MODE 3 or 4 to adequately simulate operating temperature and pressure effects on PORV operation.

SR 3.4.11.3

Verifying OPERABILITY of the safety related nitrogen supply for the Class I PORVs may be accomplished by:

- a. Isolating and venting the normal air supply, and
- b. Verifying that any leakage of the Class I backup nitrogen system is within its limits, and
- c. Operating the Class I PORVs through one complete cycle of full travel.

Operating the solenoid nitrogen control valves and check valves on the nitrogen supply system and operating the Class I PORVs through one complete cycle of full travel ensures the nitrogen backup supply for the Class I PORV operates properly when called upon. The Frequency of 18 months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate Class I PORV OPERABILITY.

SR 3.4.11.4

Not Used

REFERENCES

1. Regulatory Guide 1.32, February 1977.
 2. FSAR, Section 15.2.
 3. ASME, Code for Operation and Maintenance of Nuclear Power Plants, 1987, with 1988 Addenda, Part 10.
 4. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and generic issue 94, 'Additional Low-Temperature Overpressure for Light-Water Reactors,' Pursuant to 10CFR50.54(f)," June 25, 1990.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PRESSURE TEMPERATURE LIMITS REPORT (PTLR) provides the allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperatures during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only after temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all SI pumps and one centrifugal charging pump (CCP) incapable of injection into the RCS and isolating the accumulators.

Although not addressed in the LCO, the plant design also includes a positive displacement charging pump (PDP). Operation of a CCP or a PDP or simultaneous operation of both is controlled administratively. The pump operating combinations and limitations are discussed in B 3.4.12 ACTION B.1.

The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

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BASES

BACKGROUND
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The pressurizer has three Power Operated Relief Valves (PORVs). Two of the three PORVs are classified as safety related and are designated for LTOP pressure protection. All the PORVs are air operated. These two safety related PORVs have a nitrogen gas backup to the non-safety related air supply.

The three PORVs are the same design. The PORV that is designated as non-Class I may be used, when instrument air is available, to control RCS pressure similarly to the Class I PORVs although the non-Class I PORV does not receive an automatic open signal like the LTOP designated valves. Therefore, because no credit is taken for its operation for LTOP, continued operation with the non-Class I PORV unavailable for RCS pressure control is allowed as long as the associated block valve or non-Class I PORV can be closed to maintain the RCS pressure boundary.

In MODE 4 with the RHR loops in operation and in MODES 5 and 6, the operating RHR loop, connected to the RCS, can provide pressure relief capability through the RHR suction line relief valve. This capacity for RCS pressure relief is not assumed in the PTLR LTOP considerations and analyses and is not included in the LCO, ACTIONS, or Surveillances.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one CCP for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

Additionally, CCPs in excess of the above limitations can be momentarily capable of injection into the RCS for swapping of inservice CCPS. This condition is acceptable based on the operator's attentiveness to RCS pressure during the pump switch over and the capability of the operator to limit a pressure increase.

The LTOP System for pressure relief consists of two Class I PORVs with reduced lift settings or a depressurized RCS and an RCS vent of sufficient size. Two RCS Class I PORVs are required for redundancy. One RCS Class I PORV has adequate relieving capability to prevent overpressurization from the allowable coolant input capability.

PORV Requirements

(continued)

BASES

As designed for the LTOP System, each Class I PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation setpoint. The evolution of RHR cooldown with no

BACKGROUND
(continued)

RCP forced circulation represents a condition where variation in RCS cold leg temperatures may occur. The RCS loop 2 and 3 wide range cold leg temperature indications provide the temperature input signal. Temperature indications from these two loops were selected to constitute a good representation of the overall four loop temperatures. However, in the event that only one RHR loop is in operation, temperature indications from RCS cold legs 2 and 3 will provide indication from a RCS loop into which the cooler water from the RHR discharge is entering. All four cold leg temperature indications are in the control room and provide a loop by loop comparison for the operator.

The LTOP system is placed into service and the block valves verified to be open by procedure at a RCS pressure of about 350 psig. This is an administrative action, not required by TS. However, if LTOP has not been placed into service prior to when the RCS temperature decreases to a temperature of about 270°F, the LTOP enable alarm annunciates to alert the operator to place the LTOP system into service. Placing LTOP into service at this point is required to satisfy the LTOP Applicability requirements. Following being placed into service, LTOP will receive RCS temperature and pressure input. The PTLR LTOP pressure setpoint is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the LTOP value, and the temperature is lower than the enable temperature, a PORV is signaled to open. The two Class I PORVs operate individually with their own setpoints.

The PTLR specifies the setpoints for LTOP. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PORV opens in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure during a RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the

(continued)

BASES

flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

APPLICABLE
SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperatures above the temperature below which LTOP is required as specified in the PTLR, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At or below the limiting temperature specified in the PTLR, overpressure prevention falls to two OPERABLE RCS Class I PORVs or to a depressurized RCS and a sufficiently sized RCS vent. Each of these means has a limited overpressure relief capability.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the RCS relief valve method or the depressurized and vented RCS condition. The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

- a. Inadvertent safety injection;
- b. Charging/letdown flow mismatch;
- c. Accumulator discharge.

Heat Input Type Transients)

- a. Inadvertent actuation of pressurizer heaters;
- b. Loss of RHR cooling; or
- c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

(continued)

BASES

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

- a. Rendering all SI pumps and one CCP incapable of injection;
- b. Deactivating the accumulator discharge isolation valves in their closed positions; and

APPLICABLE
SAFETY ANALYSES
(continued)

- c. Precluding start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop and pressurizer water level is not less than 50%. LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection.

The Reference 4 analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only CCP is actuated. Thus, the LCO allows only one CCP OPERABLE during the LTOP MODES. Since neither one RCS relief valve nor the RCS vent can handle the pressure transient resulting from accumulator injection, when RCS temperature is low the LCO also requires accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

The current DCPD temperature of LTOP Applicability of 270°F was determined in agreement with NRC Branch Technical Position 5-2. This number was added to the current TS by LA 100/99.

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6), requirements by having a maximum of one CCP OPERABLE and SI actuation enabled.

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the P/T limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one CCP injecting into the RCS with the positive displacement charging pump (PDP) operating and with RCS letdown isolated. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at

(continued)

BASES

or below the derived limit ensures the Reference 1 P/T limits will be met at low temperature operation.

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron

APPLICABLE
SAFETY ANALYSES
(continued)

fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The PTLR discusses these examinations.

The failure of one Class I PORV is assumed to represent the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 2.07 square inches is capable of mitigating the allowed LTOP transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, no SI pumps and one CCP OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure. The pathway from the RCS to the vent is also considered to be passive. The vent is considered to connect directly to the RCS. If the pathway includes devices with the potential to block the pathway, these devices must be secured to avoid blocking the vent.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when RCS coolant input and pressure relief capabilities are within limits established in the LCO. Violation of this LCO could lead to the loss of low temperature overpressure mitigation capability and violation of the PTLR limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires that a maximum of zero SI pumps and one CCP (except during pump swap

(continued)

BASES

operations) be capable of injecting into the RCS, and all accumulator discharge isolation valves be closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

APPLICABLE
SAFETY ANALYSES
(continued)

- a. Two RCS Class I PORVs as follows:

A Class 1 PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

OR

- b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of ≥ 2.07 square inches.

Either of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient. The LCO is modified by a Note that permits two CCPs capable of injecting into the RCS until one hour after completion of pump swap operation.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is \leq the temperature below which LTOP is required as specified in the PTLR, in MODE 5, and in MODE 6 when the reactor vessel head is on and the vessel head closure bolts are not fully de-tensioned. RCS overpressure protection is not required in MODE 6 with the reactor vessel head closure bolts fully de-tensioned. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the limiting temperature specified in the PTLR. When the reactor vessel head is off, overpressurization cannot occur.

The PTLR provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, 3, and MODE 4 above the limiting RCS temperature specified in the PTLR.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input

(continued)

BASES

transient can cause a very rapid increase in RCS pressure when little or no time is available for operator action to mitigate the event.

The Applicability is modified by two Notes. Note 1 states that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions. Note 2 states that more than one charging pump may be capable of injection into the RCS during, and up to 1 hour after swapping charging pump operation.

ACTIONS

A.1 and B.1

With one or more SI pumps or two CCPs capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

The CCP and the PDP are capable of injecting into the RCS both operating alone or simultaneously. Their operation is limited to the conditions specified in the PTLR. The current limitations are based on RCS temperature as follows:

<u>RCS Temperature</u>	<u>Allowable Charging Pumps Capable of Injecting into the RCS.</u>
Greater than 270°F	Two CCPs AND one PDP
Less than or equal to 270°F but greater than approximately 162°F	One CCP AND one PDP
Less than or equal to approximately 162°F but greater than approximately 134°F	One CCP OR one PDP
Less than or equal to approximately 134°F	One CCP OR one PDP AND ECCS charging injection flow path isolated

If the RCS reactor vessel head is fully de-tensioned or the RCS is not intact, the above CCP and PDP limitations do not apply.

(continued)

BASES

C.1, D.1, and D.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required ACTION D.1 and Required ACTION D.2 provide two options,

ACTIONS
(continued)

either of which must be performed in the next 12 hours. By increasing the RCS temperature to > the temperature below which LTOP is required as specified in the PTLR, an accumulator pressure of 600 psig cannot exceed the P/T limits if the accumulators are fully injected. The second option to depressurize the accumulators below the P/T limits from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

E.1

In MODE 4 when any RCS cold leg temperature is \leq the temperature below which LTOP is required as specified in the PTLR, with one required RCS Class I PORV inoperable, the RCS Class I PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS Class I PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the RCS Class I PORVs is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

F.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS Class I PORVs inoperable in MODE 5 or in MODE 6 with the head on and the vessel head closure bolts not fully de-tensioned, the Completion Time to restore two valves to OPERABLE status is 24 hours.

(continued)

BASES

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS Class I PORV to protect against overpressure events.

G.1

The RCS must be depressurized and a vent must be established within 8 hours when:

ACTIONS
(continued)

- a. Both required RCS Class I PORVs are inoperable; or
- b. A Required Action and associated Completion Time of Condition A, B, D, E, or F is not met; or
- c. The LTOP System is inoperable for any reason other than Condition A, B, C, D, E, or F.

The vent must be sized ≥ 2.07 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE
REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of zero SI pumps and one CCP are verified capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and their breakers open. Verification that each accumulator is isolated is only required when accumulator isolation is required as stated in Note 1 to the Applicability.

The SI pumps and one CCP are rendered incapable of injecting into the RCS for example, through removing the power from the pumps by racking the breakers out under administrative control or by isolating the discharge of the pump by closed isolation valves with power removed from the operators or by a manual isolation valve secured in the closed position.

(continued)

BASES

An alternate method of providing low temperature overpressure protection may be employed to prevent a pump start that could result in an injection into the RCS. An inoperable pump may be energized for test or for accumulator fill provided the discharge of the pump is isolated from the RCS by closed isolation valve(s) with power removed from the valve operator(s), or by manual isolation valve(s) sealed in the closed position. The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.12.4

Not Used

SR 3.4.12.5

The RCS vent of ≥ 2.07 square inches is proven OPERABLE by verifying its open condition either:

- a. Once every 12 hours for a valve that is not locked, sealed, or secured in the open position.
- b. Once every 31 days for other vent paths (e.g., a valve that is locked, sealed, or otherwise secured in the open position. A removed pressurizer safety valve or open manway also fits this category.

Any passive vent path arrangement need only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of LCO 3.4.12.b.

SR 3.4.12.6

The Class I PORV block valve must be verified open every 72 hours to provide the flow path for each required Class I PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

(continued)

BASES

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the Class I PORV block valve remains open.

SR 3.4.12.7

Not Used

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.12.8

The Note SR states that the SR is not required to be performed until 12 hours after decreasing any RCS cold leg temperature to \leq the temperature below which LTOP is required as specified in the PTLR.

The test must be performed within 12 hours after entering the LTOP MODES. The 12 hour allowance considers the unlikelihood of a low temperature overpressure event during this time.

Following the initial 12 hour SR, while remaining in the Applicable LTOP MODE, the SR will be performed every 31 days thereafter on each required Class I PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

SR 3.4.12.9

Performance of a CHANNEL CALIBRATION on each required Class I PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. NOT USED
4. FSAR, Chapter 5.
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.

BASES

7. Generic Letter 90-06.

8. NOT USED

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE.

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

Possible leakage from a Control Rod Drive Mechanism (CRDM) canopy seal weld may be construed as either identified or unidentified LEAKAGE but not construed as pressure boundary LEAKAGE in accordance with Westinghouse letter PGE-88-622.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leak tight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The SGTR (Ref. 3) is more limiting for radiological releases at the site boundary. The radiological dose analysis assumes loss of off-site power at the time of reactor trip with no subsequent condenser cooling available. The steam generator (SG) PORV for the SG that has sustained the tube rupture is assumed to be open for 30 minutes, at which time the RCS pressure is below the lift setting of the PORV. The dose consequences resulting from the SGTR accident are within the limits defined in 10 CFR 100 (Ref. 6).

The safety analysis for RCS main loop piping for GDC-4 (Ref. 1) assumes 1 gpm unidentified leakage and monitoring per RG 1.45 (Ref. 2) are maintained (Ref. 4 and 5).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals, gaskets, or the CRDM canopy seal welds is not pressure boundary LEAKAGE. Pressure boundary leakage is defined as "non-isolable" leakage. A "non-isolable" RCS leak is one that is not capable of being isolated from the RCS using installed automatic or accessible manual valves.

(continued)

BASES

LCO
(continued)b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Identified LEAKAGE does not include LEAKAGE from portions of the Chemical and Volume Control System outside of containment that can be isolated from the RCS. LEAKAGE of this nature may be reviewed for possible impact on the Primary Coolant Sources Outside Containment Program. Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through Any One SG

The 500 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized:

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or if unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be met with the reactor at steady state operating conditions. Therefore, a Note is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, (T_{avg} changes less than 5°F per hour) power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and by the containment structure sump level and flow monitoring system. It should be noted that LEAKAGE past seals, gaskets or CRDM canopy seal welds is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The 12 hour Frequency after steady state operation has been achieved provides for those situations following a transient such that the 72 hours plus extension allowed by SR 3.0.2 would be exceeded. Under these circumstances, the SR would be due within 12 hours after steady state operation has been reestablished and every 72 hours thereafter during steady state operation. Steady state is defined as T_{avg} being changed by less than 5°F/hour.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions. This surveillance does not tie directly to any of the leakage criteria in the LCO or of the conditions; therefore failure to meet this surveillance is considered failure to meet the integrity goals of the LCO and LCO 3.0.3 applies.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
 2. Regulatory Guide 1.45, May 1973.
 3. FSAR, Section 15.
 4. FSAR, Section 3.
 5. NUREG-1601, Volume 3, November, 1984.
 6. 10 CFR 100.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leak tight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. Exceeding the leakage limit may indicate the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 4) that identified potential intersystem LOCAs as a significant contributor to the risk of core melt. A subsequent study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

PIVs are provided to isolate the RCS from the following typically connected systems:

- a. Residual Heat Removal (RHR) System;
- b. Safety Injection System; and
- c. Chemical and Volume Control System.

(continued)

BASES

BACKGROUND
(continued)

Violation of this LCO could result in continued degradation of a PIV, which could leak to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE
SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

(continued)

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1, A.2.1, and A.2.2

The flow path must be isolated by two valves. Required ACTIONS A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Required ACTION A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required ACTION A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring the RCS PIV to within limits. The 72 hours Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This time frame considers the time required to complete the ACTION and the low probability of a second valve failing during this time period.

B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and

(continued)

BASES

MODE 5 within .36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. This method results in testing each valve separately. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) O&M Code, Part 10 (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Test pressures less than 2235 psig but greater than 150 psig are allowed for valves where higher pressures would tend to diminish leakage channel opening. Observed leakage shall be adjusted for actual pressure to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one half power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been resealed. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.4.14.1 (continued)

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

Testing is not required for the RHR suction isolation valves more frequently than 18 months as these valves are motor operated with control room position indication, inadvertent opening interlocks, and system high pressure alarms.

SR 3.4.14.2

Not Used

REFERENCES

1. 10 CFR 50:2.
 2. 10 CFR 50.55a(c).
 3. 10 CFR 50, Appendix A, Section V, GDC 55.
 4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
 5. NUREG-0677, May 1980.
 6. NOT USED
 7. ASME Code for Operation and Maintenance of Nuclear Power Plants, 1987, with 1988 Addenda, Part 10.
 8. 10 CFR 50.55a(g).
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sumps used to collect unidentified LEAKAGE and the containment fan cooling unit (CFCU) condensate collection monitors are capable of detecting increases of 0.5 to 1.0 gpm in the normal flow rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

Each CFCU has an individual condensate collection monitor. The condensate from the cooling coils passes out from the CFCU to a containment sump. The condensate collection system design does not use an on-line flow monitor. The condensate drain flow can be collected, measured, and then using the elapsed time of the collection, the average flow rate can be determined. This monitoring can be done from the control room. Although multiple CFCUs may be operating, any individual CFCU condensate monitor may be employed to provide indication of the condensate flow rate.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} $\mu\text{Ci}/\text{cc}$ radioactivity for particulate monitoring and of 10^{-6} $\mu\text{Ci}/\text{cc}$ radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous

(continued)

BASES

BACKGROUND
(continued)

activities because of their sensitivities and rapid responses to RCS LEAKAGE. Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

APPLICABLE
SAFETY ANALYSES

The asymmetric loads produced by postulated breaks are the result of assumed pressure imbalance, both internal and external to the RCS. The internal asymmetric loads result from a rapid decompression that causes large transient pressure differentials across the core barrel and fuel assemblies. The external asymmetric loads result from the rapid depressurization of the annulus regions, such as the annulus between the reactor vessel and the shield wall, and cause large transient pressure differentials to act on the vessel. These differential pressure loads could damage RCS supports, core cooling equipment or core internals. This concern was first identified as Multiplant Action (MPA) D-10 and subsequently as Unresolved Safety Issue (USI) 2, "Asymmetric LOCA Loads" (Ref. 4).

The resolution of USI-2 for Westinghouse PWRs was the use of fracture mechanics technology for RCS piping > 10 inches diameter. (Ref. 5). This technology became known as leak before-break (LBB). Included within the LBB methodology was the requirement to have leak detection systems capable of detecting a 1.0 gpm leak within four hours. This leakage rate is designed to ensure that adequate margins exist to detect leaks in a timely manner during normal operating conditions. The use of the LBB methodology is described in Reference 6.

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 3).

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leak occur that could be detrimental to the safety of the unit and the public.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36 (c) (2) (ii).

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitoring systems, the particulate radioactivity monitor and either a CFCU condensate collection monitor or a gaseous radioactivity monitor provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE. In MODE 5 or 6, the temperature is to be $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

ACTIONS are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the required containment sump monitor, the required atmospheric particulate monitor, the required atmospheric gaseous monitor or the required CFCU condensate collection monitor are inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.

A.1 and A.2

With the required containment sump monitors inoperable, RCS water inventory balance, the containment atmosphere particulate radioactivity monitor, and the CFCU condensate collection monitoring system will provide indications of changes in leakage.

(continued)

BASES

ACTIONS
(continued)

Together with the atmosphere monitors, the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (near rated operating pressure with stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required sump monitoring system to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitoring system failure. This time is acceptable considering the Frequency and adequacy of the RCS water inventory balance required by required Action A.1.

B.1.1, B.1.2, and B.2

With the particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment atmosphere particulate radioactivity monitor.

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (with stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day Completion Time recognizes at least one other form of LEAKAGE detection is available.

C.1.1, C.1.2, C.2.1, and C.2.2

With the required containment atmosphere gaseous radioactivity monitor and the required CFCU condensate collection monitor inoperable, the means of detecting leakage are the containment

(continued)

BASES

ACTIONS
(continued)

sump monitoring system and the containment atmosphere particulate radioactivity monitor. This Condition does not provide all the required diverse means of leakage detection. With both gaseous containment atmosphere radioactivity monitoring and CFCU condensate monitoring instrumentation channels inoperable, alternate action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information.

The follow up Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.

D.1 and D.2

If a Required Action of Condition A, B, or C, cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

With all required monitors inoperable, (LCO a, b, and c) no means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required. With two of the three groups of leak detection monitoring not operable, the two groups will enter their respective ACTION and Completion statements. The third group provides a continued monitoring function.

SURVEILLANCE
REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitors. The check gives reasonable confidence that the channels are operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off-normal conditions.

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BASES

SURVEILLANCE -
REQUIREMENTS
(continued)

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a Channel Functional Test (CFT) on the required containment atmosphere radioactivity monitors. The test ensures that the monitors can perform their function in the desired manner including alarm functions. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3, SR 3.4.15.4, and SR 3.4.15.5

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
 2. Regulatory Guide 1.45.
 3. FSAR, Section 5.2.7.
 4. NUREG-609, "Asymmetric Blowdown Loads on PWR Primary System," 1981.
 5. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Breaks in PWR Pipe Breaks in PWR Primary Main Loops."
 6. FSAR, Section 3.6B.
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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriate fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE
SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 from LCO 3.7.18, "Secondary Specific Activity."

The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 $\mu\text{Ci/gm}$ DOSE

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 50 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of $100/E$ $\mu\text{Ci/gm}$ for gross specific activity.

The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.

The safety analysis shows the radiological consequences of an SGTR accident are within a fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131.

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

The specific iodine activity is limited to 1.0 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of $\mu\text{Ci/gm}$ equal to 100 divided by E (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on

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BASES

LCO
(continued)

DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a fraction of the allowed whole body dose.

The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq 500^{\circ}\text{F}$, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature $< 500^{\circ}\text{F}$, and in MODES 4 and 5, the offsite release of radioactivity from the affected SG in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety and relief valves.

ACTIONS

A.1 and A.2

A Note to these ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is allowed to obtain and analyze a sample. Sampling is continued to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is allowed to permit recovery, if the limit violation resulted from normal iodine spiking.

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BASES

ACTIONS
(continued)B.1 and B.2

With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply.

The change within 6 hours to MODE 3 and to reduce RCS average temperature $< 500^{\circ}\text{F}$ in 12 hours lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the affected SG to the environment in an SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions and the Completion Time of 12 hours is reasonable to reduce T_{avg} below 500°F in an orderly manner and without challenging plant systems.

C.1 and C.2

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 within 6 hours and RCS average temperature (T_{avg}) reduced to $< 500^{\circ}\text{F}$ within 12 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions and the Completion Time of 12 hours is reasonable to reduce T_{avg} below 500°F in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within two hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with $T_{\text{avg}} \geq 500^{\circ}\text{F}$. The 7 day Frequency

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BASES

SURVEILLANCE
REQUIREMENTS -
(continued)

considers the'unlikelyhood of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change $\geq 15\%$ RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide less indicative results.

SR 3.4.16.3

A radiochemical analysis for \bar{E} determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium (as defined in SR 3.4.16.3 NOTE) conditions. The \bar{E} determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for \bar{E} is the qualitative measurement of the specific activity for each radionuclide, except for radioiodines which are identified in the reactor coolant. The specific activity for these individual radionuclides shall be used in the determination of \bar{E} for the reactor coolant sample. Determination of the contributors to \bar{E} shall be based upon those energy peaks identifiable with a 95% confidence level. The Frequency of 184 days recognizes \bar{E} does not change rapidly.

This SR has been modified by a Note that indicates sampling for \bar{E} determination is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for \bar{E} is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 100.11, 1973.
 2. FSAR, Sections 15.4.3 and 15.5.20.
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Accumulators

BASES

BACKGROUND

The functions of the ECCS accumulators are to supply borated water to replace inventory in the reactor vessel during the refill phase of a loss of coolant accident (LOCA) and to provide Reactor Coolant System (RCS) makeup for a small break LOCA.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the containment atmosphere.

In the refill phase of a LOCA, which immediately follows the blowdown phase, reactor coolant inventory has vacated the core through steam flashing and spill out through the break. The core is essentially in adiabatic heatup. The balance of accumulator inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection (SI) water.

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The accumulators are passive components, since no operator or control actions are required in order for them to perform their function. Internal accumulator tank pressure is sufficient to discharge the accumulator contents to the RCS, if RCS pressure decreases below the accumulator pressure.

Each accumulator is piped into an RCS cold leg via an accumulator line and is isolated from the RCS by two check valves in series and by an open motor operated isolation valve (8808A, B, C, and D). The motor operated isolation valves are interlocked by P-11 with the pressurizer pressure measurement channels to receive an "open" signal when permissive circuit P-11 is cleared. However, before permissive circuit P-11 is reached, these valves are manually opened and their motor operator breakers are sealed open to satisfy SR 3.5.1.5. Therefore, in the event of a LOCA, accumulator actuation is passive. (Ref. 6)

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. However, if these valves were closed, they would be automatically opened as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for

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BASES

BACKGROUND
(continued)

"operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE
SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2 and 4). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

The limiting large break LOCA is a double ended guillotine break at the discharge of the reactor coolant pump. During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow with no credit taken for ECCS pump flow until an effective delay has elapsed. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break. No operator action is assumed during the blowdown stage of a large break LOCA.

The worst case small break LOCA analyses also assume a time delay before pumped flow reaches the core. For the larger range of small breaks, the SI pumps begin RCS injection, however, the increase in fuel clad temperature is terminated primarily by the accumulators, with pumped flow then providing continued cooling. As break size decreases, the accumulators and the ECCS centrifugal charging and SI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the accumulators continues to decrease. The accumulators do not discharge above the pressure of their nitrogen cover gas (595.5 to 647.5 psig.) At higher pressures the ECCS centrifugal charging pumps and SI pumps injection becomes solely responsible for terminating the temperature increase.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

This LCO helps to ensure that the following acceptance criteria established for the ECCS by 10 CFR 50.46 (Ref. 3) that are applicable for the accumulators will be met following a LOCA:

- a. Maximum fuel element cladding temperature is $\leq 2200^{\circ}\text{F}$;
- b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium-water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding cladding surrounding the plenum volume, were to react; and
- d. Core is maintained in a coolable geometry.

Since the accumulators discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.

For both the large and small break LOCA analyses, a nominal contained accumulator water volume is used. The contained water volume is the same as the deliverable volume for the accumulators, since the accumulators are emptied, once discharged. For small breaks, an increase in water volume is a peak clad temperature penalty. Depending on the NRC-approved methodology used to analyze large breaks, an increase in water volume may result in either a peak clad temperature penalty or benefit, depending on downcomer filling and subsequent spill through the break during the core reflooding portion of the transient. The analysis makes a conservative assumption with respect to ignoring or taking credit for line water volume from the accumulator to the check valve. The safety analysis assumes values of $\geq 60.8\%$ (836 cubic feet) and $\leq 72.6\%$ (864 cubic feet) as read on narrow range level instruments (not including instrument uncertainty).

The minimum boron concentration setpoint is used in the post LOCA boron concentration calculation. The calculation is performed to assure reactor subcriticality in a post LOCA environment. Of particular interest is the large break LOCA, since no credit is taken for control rod assembly insertion. A reduction below the accumulator LCO minimum boron concentration would produce a subsequent reduction in the available containment recirculation sump boron concentration for post LOCA shutdown and an increase in the sump pH. The maximum boron concentration is used in determining the cold leg to hot leg recirculation injection switchover time and minimum sump pH.

The large and small break LOCA analyses are performed at the minimum nitrogen cover pressure (595.5 psig), since sensitivity analyses have demonstrated that higher nitrogen cover pressure

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BASES

APPLICABLE
SAFETY ANALYSES
(continued)

results in a computed peak clad temperature benefit. The maximum nitrogen cover pressure limit (647.5 psig) provides margin to assure inadvertent relief valve actuation does not occur.

These analysis-assumed pressures are specified in the SRs. Volumes are shown on the control board indicators as % readings on accumulator narrow range level instruments. Adjustments to the analysis parameters for instrument inaccuracies or other reasons are applied to determine the acceptance criteria used in the plant surveillance procedures. These adjustments assure the assumed analyses parameters are maintained.

The effects on containment mass and energy releases from the accumulators are accounted for in the appropriate analyses (Refs. 2 and 4).

The accumulators satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO establishes the minimum conditions required to ensure that the accumulators are available to accomplish their core cooling safety function following a LOCA. Four accumulators are required to ensure that 100% of the contents of three of the accumulators will reach the core during a LOCA. This is consistent with the assumption that the contents of one accumulator spill through the break. If less than three accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 3) could be violated.

For an accumulator to be considered OPERABLE, the isolation valve must be fully open, power removed above 1000 psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

This LCO is only applicable at RCS pressures > 1000 psig. At pressures ≤ 1000 psig, the rate of RCS blowdown is such that the ECCS pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 3) limit of 2200°F.

In MODE 3, with RCS pressure ≤ 1000 psig, and in MODES 4, 5, and 6, the accumulator motor operated isolation valves are normally closed to isolate the accumulators from the RCS. This allows RCS cooldown and depressurization without discharging the

(continued)

BASES

accumulators into the RCS or requiring depressurization of the accumulators.

ACTIONSA.1

If the boron concentration of one accumulator is not within limits, it must be returned to within the limits within 72 hours. In this Condition, the ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analyses demonstrate that the accumulators will discharge following a large main steam line break. The impact of their discharge is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and RCS pressure reduced to ≤ 1000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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BASES

ACTIONS
(continued)

D.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE
REQUIREMENTS

SR 3.5.1.1

Each accumulator motor operated isolation valve (8808A, B, C, and D) should be verified to be fully open every 12 hours. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as in-leakage. Sampling the affected accumulator within 6 hours after a solution volume increase of 5.6% (101 gallon) narrow range indicated level will identify whether in-leakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), and the RWST has not been diluted since verifying that its boron concentration satisfies SR 3.5.4.3, because the water contained in the RWST is nominally within the accumulator boron concentration requirements as verified by SR 3.5.4.3. This is consistent with the recommendation of GL 93-05 (Ref. 5).

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator (8808A, B, C, and D) when the RCS pressure is greater than 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when RCS pressure is less than or equal to 1000 psig, thus allowing the valves to be closed to enable plant shutdown without discharging the accumulators into the RCS. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock (P-11) associated with the valves.

Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

REFERENCES

1. IEEE Standard 279-1971.
 2. FSAR, Chapter 6.
 3. 10 CFR 50.46.
 4. FSAR, Chapter 15.
 5. GL 93-05, Item 7.1.
 6. DCM S-38A
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents:

- a. Loss of coolant accident (LOCA), non-isolable coolant leakage greater than the capability of the normal charging system;
- b. Rod ejection accident;
- c. Loss of secondary coolant accident, including uncontrolled steam release or loss of feedwater; and
- d. Steam generator tube rupture (SGTR).

The addition of negative reactivity is designed primarily for the loss of secondary coolant accident where primary cooldown could add enough positive reactivity to achieve criticality and return to significant power.

The ECCS consists of three separate subsystems: centrifugal charging (high head), safety injection (SI) (intermediate head), and residual heat removal (RHR) (low head). Each subsystem consists of two redundant, 100% capacity trains. The ECCS accumulators and the Refueling Water Storage Tank (RWST) are also part of the ECCS, but are not considered part of an ECCS flow path as described by this LCO.

The ECCS components are divided into two trains, A and B. The following are the train assignments for the ECCS pumps.

Train A: RHR Pump 2	Train B: RHR Pump 1
SI Pump 1	SI Pump 2
Centrifugal Charging Pump (CCP) 1	Centrifugal Charging Pump (CCP) 2

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following the accidents described in this LCO. The major components of each subsystem are the CCPs, the RHR pumps, heat exchangers, and the SI pumps.

BACKGROUND
(continued)

Each of the three subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to

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BASES

mitigate the accident consequences. This interconnecting and redundant subsystem design provides the operators with the ability to utilize components from opposite trains to achieve the required 100% flow to the core.

There are three phases of ECCS operation following a LOCA: injection, cold leg recirculation, and hot leg recirculation. In the injection phase, water is taken from the RWST and injected into the Reactor Coolant System (RCS) through the cold legs. When sufficient water is removed from the RWST to ensure that enough boron has been added to maintain the reactor subcritical and the containment recirculation sump has enough water to supply the required net positive suction head to the RHR pumps, suction is switched to the containment recirculation sump for cold leg recirculation. After several hours, the ECCS operation is shifted to the hot leg recirculation phase to provide reverse flow through the core to backflush out the high boron concentration that could result from core boiling after a cold leg break.

During the injection phase of LOCA recovery, a suction header supplies water from the RWST to the ECCS pumps. The RWST header supplies separate piping for each subsystem. The discharge from the CCPs combines in a common header and then divides again into four supply lines, each of which feeds the injection line to one RCS cold leg. The discharge from the SI and RHR pumps divides and feeds an injection line to each of the RCS cold legs. Throttle/runout valves are set to balance the flow to the RCS. The throttle/runout valves also protect the SI and CCPs from exceeding their runout flow limits. This balance ensures sufficient flow to the core to meet the analysis assumptions following a LOCA in one of the RCS cold legs.

For LOCAs that are too small to depressurize the RCS below the shutoff head of the SI pumps, the CCPs supply water until the RCS pressure decreases below the SI pump shutoff head. During this period, the steam generators are used to provide part of the core cooling function.

During the recirculation phase of LOCA recovery, RHR pump suction is transferred to the containment recirculation sump. The RHR pumps then supply the other ECCS pumps. Initially, recirculation discharge is through the same paths as the injection phase to the cold legs. Subsequently, recirculation provides injection to both the hot and cold legs.

The centrifugal charging subsystem of the ECCS also functions to supply borated water to the reactor core following increased

BACKGROUND
(continued)

heat removal events, such as a main steam line break (MSLB). The limiting design conditions occur when the negative moderator

(continued)

BASES

temperature coefficient is highly negative, such as at the end of each cycle.

During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.

The ECCS subsystems are actuated upon receipt of an SI signal. The actuation of safeguard loads is accomplished in a programmed time sequence. If offsite power is available, the safeguard loads start after a one second sequencer delay in the programmed time sequence. If offsite power is not available, the Engineered Safety Feature (ESF) buses shed normal operating loads and are connected to the emergency diesel generators (EDGs). Safeguard loads are then actuated in the programmed time sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the time required before pumped flow is available to the core following a LOCA.

Each ECCS pump is provided with normally open miniflow lines for pump protection. The RHR miniflow isolation valves close on flow to the RCS and have a time delay to prevent them from closing until the RHR pumps are up to speed and capable of delivering fluid to the RCS. The SI pump miniflow isolation valves are closed manually from the control room prior to transfer from injection to recirculation. The CCP miniflow isolation valves are also closed manually from the control room prior to transfer from injection to recirculation.

The active ECCS components, along with the passive accumulators and the RWST covered in LCO 3.5.1, "Accumulators," and LCO 3.5.4, "Refueling Water Storage Tank (RWST)," provide the cooling water necessary to meet GDC 35 (Ref. 1).

APPLICABLE
SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 2), will be met following a LOCA:

- a. Maximum fuel element cladding temperature is $\leq 2200^{\circ}\text{F}$;
- b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;
- c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

APPLICABLE
SAFETY ANALYSIS

- d. Core is maintained in a coolable geometry; and

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BASES

(continued)

- e. Adequate long term core cooling capability is maintained.

The LCO also limits the potential for a post-trip return to power following an MSLB event and ensures that containment temperature limits are met.

Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4). This event establishes the requirement to limit runout flow for the ECCS pumps, as well as the maximum response time for their actuation. The centrifugal charging pumps and SI pumps are credited in the injection phase for mitigation of a small break LOCA event. This event establishes the flow and discharge head for the design point of the CCPs. The SGTR and MSLB events also credit the CCPs. The OPERABILITY requirements for the ECCS are based on the following LOCA analysis assumptions:

- a. A large break LOCA event, with loss of offsite power and a single failure disabling one RHR pump (all EDG trains are assumed to operate due to requirements for modeling full active containment heat removal system operation); and
- b. A small break LOCA event, with a loss of offsite power and a single failure disabling one ECCS train.

During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the containment. The nuclear reaction is terminated either by moderator voiding during large breaks or control rod insertion for small breaks. Following depressurization, emergency cooling water is injected into the cold legs, flows into the downcomer, fills the lower plenum, and refloods the core.

The effects on containment mass and energy releases are accounted for in appropriate analyses (Refs. 3 and 4). The LCO ensures that an ECCS train will deliver sufficient water to match boiloff rates soon enough to minimize the consequences of the core being uncovered following a large break LOCA. It also ensures that the centrifugal charging and SI pumps will deliver sufficient water and boron during a small break LOCA to maintain core subcriticality. For smaller break LOCAs, the centrifugal charging pump delivers sufficient fluid to maintain RCS inventory. For a small break LOCA, the steam generators continue to serve as the heat sink, providing part of the required core cooling.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available.

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BASES

assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal. During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold legs. The ECCS suction is manually transferred to the containment recirculation sump to place the system in the recirculation mode of operation to supply its flow to the RCS hot and cold legs. During the recirculation operation, the RHR pumps provide suction to the charging and SI pumps.

During recirculation operation, the flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

As indicated in Note 1, the SI flow paths may be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room and a single active failure (Ref. 7) is not assumed coincident with this testing. Therefore the ECCS trains are considered OPERABLE during this isolation.

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS - Shutdown."

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant

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BASES

Circulation-High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available (capable of injection into the RCS, if actuated), the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their safety function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available. (i.e. minimum of one OPERABLE CCP, SI, and RHR pump and applicable flow paths capable of drawing from the RWST and injecting into the RCS cold legs). This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

The intent of this Condition to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available applies to both the injection mode and the recirculation mode.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR cross-tie valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

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BASES

ACTIONS
(continued)

Opening the containment recirculation sump access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered (Ref. 9.)

B.1 and B.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.5.2.1

Verification of proper valve position ensures that the flow paths from the RWST containment recirculation sump, and spray additive tank to the RCS are maintained. Valve position is the concern and not indicated position in the control room. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. The surveillance can be satisfied using indicated position in the control room but may also be satisfied using local observation. These valves are of the type, described in References 6 and 7, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12 hour frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely. As noted in LCO Note 1, both SI pump flow paths may each be isolated for two hours in MODE 3 by closure of one or more of these valves to perform pressure isolation valve testing.

In addition to the valves listed in SR 3.5.2.1, there are other ECCS related valves that must be appropriately positioned. Improper valve position can affect the ECCS performance required to meet the analysis assumptions. These valves are identified in plant documents and are listed in the following table.

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SURVEILLANCE
REQUIREMENTS
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ECCS Valve Position Table

Valve Number	Valve Function	Required Valve Position	MODES
8105	CCP 1 and 2 Recirc Line Isolation	Open	1, 2, 3
8106	CCP 1 and 2 Recirc Line Isolation	Open	1, 2, 3
8716A	RHR Cross-tie Line	Open	1, 2, 3
8716B	RHR Cross-tie Line	Open	1, 2, 3
9003A	RHR to Containment Spray	Closed	1, 2, 3
9003B	RHR to Containment Spray	Closed	1, 2, 3
8804A	RHR to CCP	Closed	1, 2, 3
8804B	RHR to SI Pump	Closed	1, 2, 3
8741	RHR to RWST - Manual Valve	Closed	1, 2, 3
SI-1	RWST to ECCS - Manual Valve	Open	1, 2, 3, 4
8923A*	Train "A" SI Pump Suction Valve	Open	1, 2, 3

* Valve can be closed, but not when RHR Train "A" (containing RHR pump 2) is out of service. Closing this valve with RHR Train "A" out of service would result in both trains of ECCS being inoperable due to the ECCS piping configuration.

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a non-accident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

With the exception of the operating CCP, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of non-condensable gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

The intent of the SR is to assure the ECCS piping is full of water. Different means of verification, as alternates to venting the accessible system high points, can be employed to provide this assurance.

SR 3.5.2.4

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by the ASME Code. (Ref. 8) This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is within the performance assumed in the plant safety analysis. SRs are specified in the applicable portions of the Inservice Testing Program, which encompasses Part 6 of the ASME Code for Operation and Maintenance of Nuclear Power Plants. (Ref. 8). This section of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

The correct position of throttle/runout valves in the ECCS flow paths is necessary for proper ECCS performance. These manual throttle/runout valves are positioned during flow balancing and have mechanical locks and seals to ensure that the proper positioning for restricted flow to a ruptured cold leg is maintained. The verification of proper position of a throttle/runout valve can be accomplished by confirming the seals have not been altered since the last performance of the flow balance test. Restricting the flow to a ruptured cold leg ensures that the other cold legs receive at least the required minimum flow. The 18 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

Periodic inspections of the containment recirculation sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

Opening the containment recirculation sump access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered. (Ref. 9)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR, Sections 6.3 and 7.3.
4. FSAR, Chapter 15. "Accident Analysis."

(continued)

BASES

5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
 6. IE Information Notice No. 87-01.
 7. BTP EICSB-18, Application of the Single Failure Criteria to Manually-Controlled Electrically-Operated Valves.
 8. ASME/ANSI OM-1987, "Operational Maintenance of Nuclear Power Plants", including OM-a-1988 addenda, Part 6, "Inservice Testing of Pumps in Light Water Reactor Power Plants:," and part 10, "Inservice Testing of Valves in Light Water Reactor Power Plants."
 9. NRC letter to PG&E, EA 89-241, April 5, 1990; CHRON 148598.
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS-Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2, and subsequently transferring RHR pump suction to the containment recirculation sump.

APPLICABLE
SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable core reactivity and the lower heat removal conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuations are not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA (Ref. 1.)

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation.

The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains (as defined for MODE 4) is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment recirculation sump.

(continued)

BASES

LCO

(continued)

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS charging and RHR pumps and their respective supply headers to each of the four cold legs. In the long term, this flow path may be switched to take its supply from the containment recirculation sump and to deliver its flow to the RCS hot and cold legs.

This LCO is modified by a Note that allows an RHR train to be considered OPERABLE during system alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS high head and low head train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

(continued)

BASES

ACTIONS
(continued)

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

Opening the containment recirculation sump access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered. (Ref. 9)

B.1

With no ECCS centrifugal charging subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour, Completion Time to restore at least one ECCS centrifugal subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

SURVEILLANCE
REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply.

REFERENCES

1. Abnormal Response Guideline, ARG- 2, Rev. 0, Feb. 28, 1992.

Note: The applicable references from BASES 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions (boration flow path), to the refueling cavity during refueling, and to the ECCS and the Containment Spray (CS) System during accident conditions.

The RWST supplies both trains of the ECCS through one header and both trains of the CS System through a separate supply header during the injection phase of a loss of coolant (LOCA) recovery. Motor-operated isolation valves in each sub-system header isolate the RWST from the ECCS and from the CS System once the RWST is no longer supplying flow to these systems.

Use of a single RWST to supply both trains of the ECCS and CS Systems is acceptable since the RWST is a passive component, and a passive failure is not assumed to occur coincidentally with a Design Basis Accident (DBA).

During normal plant operation in MODES 1, 2, and 3, the Safety Injection (SI) and Residual Heat Removal (RHR) pumps are aligned to take suction from the RWST. The Centrifugal Charging Pumps (CCPs) operate during normal plant operation with their suction aligned to the Volume Control Tank (VCT). The switchover from normal operation to the injection phase of ECCS operation requires auto-transfer of the CCP suction from the CVCS VCT to the RWST. The CS pumps suction is aligned to the RWST with closed motor operated discharge valves which open on a CS signal.

When the suction for the RHR pumps is transferred to the containment recirculation sump, the RWST must be isolated from ECCS and CS system. The isolation prevents flow of containment recirculation sump water into the RWST. Flow of containment water into the RWST could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the RHR pumps due to loss of containment recirculation sump inventory.

The reactivity control systems are available to the operators to ensure that negative reactivity is available during each mode of plant operation. This system is not an automatic accident mitigation system, but is used under operator control if needed to increase the Reactor Coolant System (RCS) boration concentration. The sources of borated water are the boric acid storage tanks in the CVCS and the RWST. The RWST source of borated water is available as an alternate source to the boric acid storage tanks. RWST water can be used in the event of

(continued)

BASES

BACKGROUND
(continued)

abnormal conditions, including single active failure events that may impair the function of the boric acid storage tank source of borated water of the CVCS. The boration subsystem provides the means to meet one of the functional requirements of the CVCS, i.e., to control the neutron absorber (boron) concentration in the RCS and to help maintain the SHUTDOWN MARGIN (SDM).

The LCO ensures that:

- a. The RWST contains a sufficient volume at an acceptable boron concentration and temperature to support the ECCS and CS systems during the injection phase;
- b. Sufficient water volume exists in the containment recirculation sump to support continued operation of the ECCS System pumps at the time of transfer to the recirculation mode of cooling;
- c. The reactor remains subcritical following a LOCA.

APPLICABLE
SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and CS System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating"; B 3.5.3, "ECCS - Shutdown"; and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

Any event that results in SI initiation, including inadvertent ECCS actuation, results in delivery of RWST water to the RCS. However, the events for which the RWST parameters provide mitigation or are limiting are large-break LOCA and steam line break. Feedwater line break and steam generator tube rupture (SGTR) also involve SI but the RWST parameters are less significant to the analysis results. RWST boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a non-limiting event and the results are very insensitive to boron concentrations. The effect of these RWST parameters on large-break LOCA, main steam line break, feedwater line break, and SGTR are discussed below:

(continued)

BASES

SAFETY ANALYSES
(continued)LOCA

Volume

Insufficient water in the RWST could result in insufficient borated water inventory in the containment recirculation sump when the transfer to the recirculation phase occurs. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is less than the total volume contained since, due to the design of the tank, the ECCS suction nozzle elevation is above the bottom of the tank, so more water can be contained than can be delivered. The contained water volume limit includes an allowance for water not usable because of tank discharge location or other physical characteristics.

Boration

During accident conditions, the RWST provides a source of borated water to the ECCS and CS System pumps. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment. The minimum boron concentration limit ensures that the spray and the containment recirculation sump solutions, after mixing with the sodium hydroxide from the spray additive tank, will not exceed the maximum pH values. The maximum boron concentration limit ensures that the containment recirculation sump solution will not be less than the minimum pH requirement. The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Diablo Canyon FSAR Update. These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses

For a large-break LOCA analysis, the RWST minimum contained water volume of 400,000 gallons (81.5% indicated level uncorrected for uncertainty) and the lower boron concentration limit of 2300 ppm are used to compute the post-LOCA sump boron concentration necessary to assure subcriticality. The large-break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration of 2500 ppm is used to determine the maximum allowable time to initiate hot leg recirculation following a LOCA. The purpose of initiating hot leg recirculation is to avoid boron precipitation in the core following the accident when the break is in the cold leg.

The use of minimum containment backpressure in the LOCA analysis results in a conservative calculation of Peak Clad Temperature (PCT). The basis for this conclusion is the effect that the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

containment pressure has on the core reflood rate. A lower containment pressure has the effect of reducing the density of the steam exiting the break, which increases the differential pressure provided by the downcomer head (this phenomena is sometimes referred to as steam binding). Thus, a higher downcomer mixture level is required to maintain the same reflood rate as before. The additional time required to establish the downcomer head translates into a reduction in the reflood rate in the core. When the downcomer has completely filled, the equilibrium reflood rate for the low containment pressure case would be less than that calculated for a high containment pressure case. This reduction in reflood rate results in a reduction in heat transfer and ultimately an increase in the calculated PCT. Thus, the regulations require that a low containment pressure be calculated in the large-break LOCA analysis.

In the ECCS analysis, the CS temperature is assumed to be equal to the RWST minimum temperature limit of 35°F. If the minimum temperature limit is violated, the CS further reduces containment pressure, which decreases the core reflood as explained in the preceding paragraph. For the containment response following a MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

Steam Line and Feedwater Line Breaks

Volume

RWST volume is not an explicit assumption in other than LOCA events since the required volume for those events is much less than that required by LOCA.

Boration

The minimum RWST solution boron concentration is an explicit assumption in the MSLB analysis to ensure the required shutdown capability. Since DCCP no longer uses the boron injection tank, the minimum boron concentration limit is an important assumption in ensuring the required shutdown capability. For the containment response following an MSLB, the lower limit on boron concentration is used to maximize the total energy release to containment.

Feedwater line break results in high temperature/high pressure in the RCS. There is very little RWST water injected due to the high pressure. Also, the analysis results are not affected by the negative reactivity provided by RWST water. Therefore, RWST boron concentration is not a consideration for the feedwater line break.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)Steam Generator Tube Rupture (SGTR)

Volume

The RWST volume needed in response to a SGTR is not an explicit assumption since the required volume is much less than that required by a LOCA.

Boration

Borated RWST water will be injected into the RCS for a SGTR event. The insertion of the control rods and the negative reactivity provided by the injected RWST solution provides sufficient SDM during the initial recovery operations. One of the initial operator recovery actions for this event is to equalize the RCS pressure and the faulted steam generator pressure to minimize or stop the primary-to-secondary tube rupture flow and terminate safety injection. Further RCS boration will be initiated by the operator by manual makeup to the RCS.

The RWST satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii)..

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment recirculation sump to support ECCS pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and CS System OPERABILITY requirements. Since both the ECCS and the CS Systems must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level."

(continued)

BASES

ACTIONS

A.1

With RWST boron concentration or borated water temperature* not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the CS System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

DCPP does not have an upper limit for RWST borated water temperature. An upper limit would typically be about 100°F. The coastal weather at the DCPP site is moderated by the Pacific Ocean and historically does not exceed 100°F. A requirement for a high temperature limit would therefore not be of value.

- * The requirement for RWST temperature is to be greater than or equal to the minimum required temperature. The expression "within the required limits", applied to RWST temperature is satisfied when the temperature is greater than or equal to the minimum.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the CS System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains and that borated water volume can be restored more rapidly than boron concentration or temperature.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and

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BASES

ACTIONS
(continued)

to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be above the minimum assumed in the accident analyses. This Frequency is sufficient to identify a temperature change that would approach the limit and has been shown to be acceptable through operating experience.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperature is above the minimum temperature for the RWST. With ambient air temperature above the minimum temperature, the RWST temperature should not exceed the limit.

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for ECCS injection and CS System pump operation and to support continued ECCS on recirculation. Since the RWST volume is normally stable and the contained volume required is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

The analysis assumed 400,000 gallons (81.5% of indicated range) is used in the TS Surveillance and is shown on the control board indicators. Adjustments to the analysis parameters for instrument inaccuracies or other reasons are applied to determine the acceptance criteria used in the plant surveillance procedures. These adjustments assure the assumed analyses parameters are maintained.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR, Chapter 6 and Chapter 15.
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B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.5 Seal Injection Flow

BASES

BACKGROUND

This LCO is applicable because the CCPs are utilized for high head safety injection (SI). The function of the seal injection throttle valves during an accident is similar to the function of the ECCS throttle valves in that each restricts flow from the CCP pump header to the Reactor Coolant System (RCS).

The restriction on reactor coolant pump (RCP) seal injection flow limits the amount of ECCS flow that would be diverted from the injection path following an accident. This limit is based on safety analysis assumptions that are required because RCP seal injection flow is not isolated during SI. Note the LCO flow limit is applicable when the ECCS system is in its injection configuration. In the normal operation configuration the indicated seal injection flow may be above the LCO limit with no impact on the ECCS flows.

APPLICABLE
SAFETY ANALYSES

All ECCS subsystems are taken credit for in the large break loss of coolant accident (LOCA) at full power (Ref. 1). The LOCA analysis establishes the minimum flow for the ECCS pumps while the inadvertent SI and the SGTR analyses establish the maximum flow for the ECCS pumps. The CCPs are also credited in the small break LOCA analysis. The small break analysis establishes the flow and discharge head requirements for CCPs performance. The SGTR and main steam line break event analyses also credit the CCPs but are not limiting in their requirements. Reference to these analyses is made in assessing changes to the Seal Injection System for evaluation of their effects in relation to the acceptance limits in these analyses.

This LCO ensures that total seal injection flow of ≤ 40 gpm, with RCS pressure ≥ 2215 psig and ≤ 2255 psig and charging flow control valve full open, will be sufficient for RCP seal integrity but limited so that the ECCS trains will be capable of delivering sufficient water to match boiloff rates soon enough to minimize uncovering of the core following a large LOCA. It also ensures that the CCPs will deliver sufficient water for a small LOCA and sufficient boron to maintain the core subcritical. For smaller LOCAs, the charging pumps alone deliver sufficient fluid to overcome the loss and maintain RCS inventory.

Seal injection flow satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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BASES

LCO

The intent of the LCO limit on seal injection flow is to make sure that flow through the RCP seal water injection line is low enough to ensure that sufficient centrifugal charging pump injection flow is directed to the RCS via the cold legs (Ref. 1). This is accomplished by limiting the line resistance in the RCP seal injection lines to a value consistent with the assumptions in the accident analysis.

The 40 gpm identified in the LCO is not an absolute flow limit, but rather a flow limit through the RCP seal injection line that is assumed in the accident analyses initial conditions when the ECCS systems are aligned in the injection mode following a LOCA. This flow value correlates to a line resistance in the seal injection flow path that is used in the accident analyses ECCS performance. Thus, the line resistance is the parameter which is controlled to ensure that the ECCS alignment is maintained consistent with the accident analysis assumptions. Charging flow control valve, FCV-128 full open is a test condition and is not indicative of normal operation. Consequently, during normal plant operation, it is possible to have the indicated total seal injection flow greater than 40 gpm while still being within the LCO because during normal plant operation, the ECCS system is not in post accident alignment.

In order to establish the proper flow line resistance, the seal injection flow path differential pressure and flow are measured. The line resistance is then determined with the RCS pressure within normal limits and the CCP flow control valve fully open.

A reduction in RCS pressure, with no concurrent decrease in CCP discharge header pressure, would result in more flow being discharged through the RCP seal injection line than at normal RCS operating pressure. The RCP seal injection valve settings established at the prescribed RCS pressure result in a conservative valve position should RCS pressure decrease. The additional modifier of this LCO, the charging flow control valve being full open, is consistent with the air operated valve assumed to fail open for the accident condition.

With the RCS pressure and control valve position as specified by the LCO, a line resistance is established which assures that the seal injection line resistance is consistent with the analysis assumptions. This limit assures that when the RCS depressurizes following a LOCA and the flow to the pump seals increases, the resulting flow to the seals will be less than the limit assumed in the accident analysis.

APPLICABILITY In MODES 1, 2, and 3, the seal injection flow limit is dictated

(continued)

BASES

by ECCS flow requirements, which are specified for MODES 1, 2, 3, and 4. The seal injection flow limit is not applicable for MODE 4 and lower, however, because high seal injection flow is less critical as a result of the lower initial RCS pressure and decay heat removal requirements in MODE 4. Therefore, RCP seal injection flow must be limited in MODES 1, 2, and 3 to ensure adequate ECCS performance.

ACTIONS

A.1 and A.2

With the seal injection flow exceeding its limit, the amount of charging flow available for ECCS injection to the RCS may be reduced. Under this Condition, action must be taken to restore the seal injection flow to below its limit. Required Action A.1 ensures that within 4 hours the remaining available ECCS charging flow (without assuming an additional failure) is $\geq 100\%$ of the assumed post-LOCA charging flow. 100% flow capability may be verified by assuring both CCPs are OPERABLE. Required Action A.2 then allows the operator 72 hours from the time the flow is known to be above the limit but still allowing 100% of the assumed post-LOCA ECCS charging flow, to correctly position the manual valves and thus be in compliance with the accident analysis. The Completion Time minimizes the potential exposure of the plant to a LOCA with insufficient injection flow and provides a reasonable time to restore seal injection flow within limits. This time is consistent with the Completion Times for ECCS in 3.5.2, ACTION A.

B.1 and B.2

When the Required Actions cannot be completed within the required Completion Time, a controlled shutdown must be initiated. The Completion Time of 6 hours for reaching MODE 3 from MODE 1 is a reasonable time for a controlled shutdown, based on operating experience and normal cooldown rates, and does not challenge plant safety systems or operators. Continuing the plant shutdown begun in Required Action B.1, an additional 6 hours is a reasonable time, based on operating experience and normal cooldown rates, to reach MODE 4, where this LCO is no longer applicable.

SURVEILLANCE
REQUIREMENTS

SR 3.5.5.1

Verification every 31 days that the manual seal injection throttle valves are adjusted to give a flow below the limit ensures proper manual seal injection throttle valve position, and hence, proper seal injection flow, is maintained. The Frequency of 31 days is based on engineering judgment and is consistent with other ECCS valve Surveillance Frequencies. The Frequency has proven to be acceptable through operating experience.

(continued)

BASES

As noted, the Surveillance is to be completed within 4 hours after the RCS pressure has stabilized within the specified pressure limits. The RCS pressure requirement is specified since this configuration will produce the required pressure conditions necessary to assure that the manual valves are set correctly. The exception is limited to 4 hours to ensure that the Surveillance is timely.

REFERENCES

1. FSAR, Chapter 6 and Chapter 15.
 2. 10 CFR 50.46.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat with a reactor cavity pit projection, and a hemispherical dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The exterior shell and concrete structure around the reactor vessel (crane wall and bio-shield wall) is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. The steel liner additionally provides support and anchorage for safety related piping and electrical raceway. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions. The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

- a. All penetrations required to be closed during accident conditions are either:
 1. capable of being closed by an OPERABLE automatic containment isolation system, or
 2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves;"
- b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";
- c. All equipment hatches are closed; and
- d. The sealing mechanism associated with a penetration (e.g. welds, bellows, or O-rings) is OPERABLE.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or a fuel handling accident. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a : the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.10% of containment air weight per day in the safety analysis at $P_a = 47$ psig (Ref. 3).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0 L_a$, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, applicable Containment Leakage Rate Testing Program leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and containment purge supply and exhaust, hydrogen purge, and containment pressure/vacuum relief valves with resilient seals (LCO 3.6.3) result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of $1.0 L_a$.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5

(continued)

BASES

to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTSSR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements as specified in the Containment Leakage Rate Testing Program which is consistent with Reg Guide 1.163, 1995, and the requirements of 10 CFR 50, Appendix J, Option B (Ref. 1). Failure to meet air lock leakage limits specified in LCO 3.6.2 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $< 0.6 L_a$ for combined Type B and C leakage following an outage or shutdown that included Type B and C testing only, and $\leq 0.75 L_a$ for overall Type A leakage following an outage or shutdown that included Type A testing. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0 L_a$. At $\leq 1.0 L_a$, the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by 10CFR50, App J, Option B. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

BASES

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
 2. FSAR, Chapter 15.
 3. FSAR, Section 6.2.
 4. Regulatory Guide 1.163 (September 1995).
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

There are two containment airlocks. The personnel air lock is nominally a right circular cylinder, approximately 9 ft in diameter, with a door at each end. The emergency air lock is approximately 5 ft inside diameter with a 2 ft 6 in door at each end. On both air locks, doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide control room indication of door position.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the safety analyses.

APPLICABLE SAFETY ANALYSIS.

In Mode 1, 2, 3, and 4, the DBA that results in a release of radioactive material within containment is a loss of coolant accident (Ref. 2). In the analysis of this accident, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as $L_a = 0.1\%$ of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure $P_a = 47.0$ psig following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS
(continued)

The containment air locks satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into or exit from containment.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

(continued)

BASES

ACTIONS
(continued)

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the limit for the air lock then the leakage must be evaluated for its effect on the overall containment leakage rate. Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment" if the overall containment leakage limits are exceeded.

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTION of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the

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BASES

ACTIONS
(continued)

Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. The air lock operability leakage limit is 0.05 L_s and is considered part of the type B and C leakage and therefore subject to the containment inoperability limit of >1.0 L_s under LCO 3.6.1. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock

BASES (continued)

ACTIONS
(continued)

have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established within the Containment Leakage Rate Testing Program. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is also required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria of the Containment

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BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

Leakage Rate Testing Program, which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. This test is only required to be performed every 24 months. The 24 month Frequency is based on avoiding the loss of containment OPERABILITY if the Surveillance were performed with the reactor at power and on engineering judgement.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
 3. FSAR, Section 3.8, 6.2, and 15.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the containment purge supply and exhaust valves, and containment pressure/vacuum relief isolation valves receive a Containment Ventilation Isolation (CVI) signal on a containment high radiation condition. In addition to these large valves, the containment gas and particulate radiation monitor penetrations also isolate upon receipt of a CVI signal. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analyses. Therefore, the

(continued)

BASES

BACKGROUND
(continued)

OPERABILITY requirements provide assurance that the containment function assumed in the safety analyses will be maintained.

Containment Purge System (48 inch purge valves)

The Containment Purge System operates to supply outside air into the containment for ventilation and cooling or heating needed for prolonged containment access following a shutdown and during refueling. The system may also be used to reduce the concentration of noble gases within containment prior to and during personnel access. The supply and exhaust lines each contain two isolation valves. The 48 inch Containment Purge valves are qualified for automatic closure from their open position under DBA conditions. Therefore, the 48 inch Containment Purge supply and exhaust isolation valves must be blocked to prevent opening more than 80° in MODES 1, 2, 3, and 4 to ensure closure within 2 seconds under DBA conditions (in order to support the required containment ventilation isolation time) and to ensure that the containment boundary is maintained. These valves may be opened as necessary to:

- a. Reduce noble gases within containment prior to and during personnel access, and
- b. Mitigate the effects of controller leakage and other sources which may effect the habitability of the containment for personnel entry.

Operation in Modes 1, 2, 3, or 4 with the 48-inch purge valves or the 12-inch vacuum/pressure relief valves open providing a flow path is limited to no more than 200 hours per calendar year.

Containment Pressure/Vacuum Relief (12 inch isolation valves)

The Containment Pressure Relief valves are operated as necessary to reduce the concentration of noble gases within containment prior to and during personnel access, and equalize containment internal and external pressures.

Since the 12 inch Containment Pressure/Vacuum Relief valves are designed to meet the requirements for automatic containment isolation within 5 seconds if mechanical blocks are installed to prevent opening more than 50°, these valves may be opened as needed in MODES 1, 2, 3, and 4.

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BASES

APPLICABLE
SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBA that result in a release of radioactive material within containment in Modes 1, 2, 3, or 4 is a loss of coolant accident (LOCA) (Ref. 1). In the analyses for this accident, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including the Containment Purge, and Containment Vacuum/pressure Relief valves) are minimized. If the 48 inch Containment Purge supply and exhaust valves close within 2 seconds and the 12 inch pressure/vacuum relief valves close within 5 seconds after the DBA initiation, the safety analysis shows that offsite dose release will be less than 10CFR100 guidelines.

The LOCA offsite dose analysis assumes leakage from the containment at a maximum leak rate of 0.10 percent of the containment volume per day for the first 24 hours, and at 0.05 percent of the containment volume per day for the duration of the accident.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the 48 inch Containment Purge supply and exhaust and the 12 inch Containment Pressure/Vacuum Relief valves. Two valves in series provide assurance that the flow paths can be isolated even if a single failure occurred. The inboard and outboard isolation valves are provided with diverse power sources and are pneumatically operated spring closed valves that will fail closed on the loss of power or air.

The 48 inch Containment Purge supply and exhaust and 12 inch Containment Pressure/Vacuum Relief valves are able to close in the environment following a LOCA. Therefore, each of the Containment Purge supply and exhaust and Containment Vacuum/pressure Relief valves may be opened to provide a flow path. The 48 inch Containment Purge supply and exhaust

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(Continued)

valves and/or 12-inch vacuum/pressure relief valves may be open no more than 200 hours per calendar year while in MODES 1, 2, 3, and 4. Additionally, only two of the three flow paths (containment purge supply and exhaust, and containment vacuum/pressure relief) may be open at one time. The system is designed to preclude a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA. The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 48 inch Containment Purge supply and exhaust valves and the Pressure/Vacuum Relief valves must have blocks installed to prevent full opening. These blocked valves also actuate on an automatic isolation signal. The valves covered by this LCO are listed along with their associated stroke times in Plant Procedure AD13.DC1, Attachment 7.10 (Ref. 5).

Normally closed passive containment isolation valves/devices are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact.

Containment Purge supply and exhaust valves, and Containment Pressure/Vacuum Relief valves with resilient seals must meet additional leakage rate surveillance frequency requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment."

This LCO provides assurance that the containment isolation valves will perform their designed safety function to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

The LCO is modified by a Note stating that the Main Steam Safety Valves, Main Steam Isolation Valves, Feedwater

BASES (continued)

LCO
(continued)

Isolation Valves, and Atmospheric Relief Valves are not addressed in this LCO. These penetration flow paths credit the steam generators and piping inside containment as a containment isolation barrier (i.e., closed system). These valves are addressed by LCO 3.7.1 "Main Steam Safety Valves (MSSVs)", LCO 3.7.2 "Main Steam Isolation Valves (MSIVs)", LCO 3.7.3 "Feedwater Isolation Valves (FIVs) and Associated Bypass Valves", and LCO 3.7.4 "Atmospheric Relief Valves (ARVs)" which provide the appropriate Required Actions in the event these valves are inoperable.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths that are normally isolated by locked or sealed closed valves or valves that do not receive a containment isolation signal, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. This Note also limits operation of the normally isolated Containment Supply and Exhaust valves (2 penetration flow paths) and the Vacuum/pressure Relief valves (1 penetration flow path) to less than 200 hours per calendar year with no more than 2 of 3 penetration flow paths open at one time.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent

(continued)

BASES (continued)

ACTIONS
(continued)

Condition entry and application of associated Required Actions:

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event the containment isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths requiring isolation following a DBA is inoperable the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated isolation valve, a closed manual valve (this includes power operated valves with power removed), a blind flange, and a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification that those isolation devices outside containment and capable of being mispositioned are in the correct

(continued)

BASES (continued)

ACTIONS
(continued)

position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with only one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by a Note 1 that applies to isolation devices located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices once they have been verified to be in the proper position, is small.

A second Note has been added to Required Action A.2 to provide clarification that the action to periodically verify the affected penetration flow path is isolated does not apply to manual valves and blind flanges that are locked, sealed, or otherwise secured. This is acceptable since these were verified to be in the correct position prior to locking, sealing, or securing.

B.1

With two containment isolation valves in one or more penetration flow paths requiring isolation following a DBA inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and

(continued)

BASES (continued)

ACTIONS
(continued)

de-activated automatic valve, a closed manual valve (this includes power operated valves with power removed), and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths requiring isolation following a DBA with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve (this includes power operated valves with power removed), and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4 (See FSAR Table 6.2-39, GDC-57 valves). In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of

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BASES (continued)

ACTIONS
(continued)

containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low. Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Reference 3. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by a Note 1 that applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

A second Note has been added to Required Action C.2 to provide clarification that the action to periodically verify the affected penetration flow path is isolated does not apply to manual valves and blind flanges that are locked, sealed, or otherwise secured. This is acceptable since these were verified to be in the correct position prior to locking, sealing, or securing.

D.1, D.2, and D.3

In the event one or more Containment Purge supply and exhaust, or Containment Pressure/Vacuum Relief isolation valves in one or more penetration flow paths are not within leakage limits, leakage must be reduced to within limits, or the affected penetration flow path must be isolated. For this Action, the leakage limit is as specified under the Leakage Rate Testing Program and exceeding this limit would require evaluation per Note 4 under LCO 3.6.3. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, closed manual valve (this includes power operated valves with power removed), or blind flange. A Containment Purge supply and exhaust, or Containment Pressure/Vacuum Relief isolation valve with

(continued)

BASES (continued)

ACTIONS
(continued)

resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.7. The specified Completion Time is reasonable, considering that one valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment leakrate following an accident, will not exceed the limit assumed in the offsite dose analysis. This Required Action does not require any testing or valve manipulation. Rather, it involves verification that those isolation devices outside containment capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

For the Containment Purge supply and exhaust, or Containment Pressure/Vacuum Relief isolation valve with resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.3.7 must be performed at least once every 92 days. This assures that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase beyond the limits during the time the penetration is isolated. The normal Frequency for SR 3.6.3.7, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 4). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per 92 days was chosen and has been shown to be acceptable based on operating experience.

E.1 and E.2

If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the

(continued)

BASES (continued)

ACTIONS
(continued)

required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.1

Not Used

SR 3.6.3.2

This SR ensures that the 48 inch Containment Purge supply and exhaust and the 12 inch Containment Pressure/Vacuum Relief valves are closed as required or, if open, open for an allowable reason. If a purge or pressure relief valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the Containment Purge supply and exhaust or Containment Pressure Relief valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The Containment Purge supply and exhaust or Containment Pressure/Vacuum Relief valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR

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BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

Verifying that the isolation time of each automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS
(Continued)

that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.6

Not Used

SR 3.6.3.7

For Containment Purge supply and exhaust, and Containment Pressure/Vacuum Relief valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining these penetrations leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 4).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

The leak rate acceptance criteria for the containment purge supply and exhaust, and containment pressure/vacuum relief valves are in accordance with the Containment Leakage Rate Testing Program.

SR 3.6.3.8

Automatic containment isolation valves close on a Phase A, Phase B, or containment ventilation isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.9

Not Used

SR 3.6.3.10

Verifying that each 12 inch containment pressure/vacuum relief valve is blocked to restrict opening to $\leq 50^\circ$ is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the containment pressure/vacuum relief valves must close to maintain containment leakage within the values assumed in the accident analysis. The 18 month Frequency is appropriate because the blocking devices are not typically removed except during maintenance.

SR 3.6.3.11

Not Used

REFERENCES

1. FSAR, Section 15.
2. FSAR, Section 6.2.
3. Standard Review Plan 6.2.4.
4. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."
5. Diablo Canyon Power Plant Administrative Procedure, AD13.DC1, Attachment 7.10

(continued)

BASES

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE
SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer modeled pressure transients. The worst case SLB (MSLB at 30% power) generates the greatest mass and energy release rate. Thus, the MSLB event bounds the LOCA event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was 16 psia (1.3 psig). This resulted in a maximum peak pressure from a MSLB of 42.25 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a , results from the limiting MSLB at 30% power. The maximum containment pressure resulting from the worst case MSLB, 42.25 psig, does not exceed the containment design pressure, 47 psig.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The containment was also designed for an external pressure load equivalent to -3.5 psig. The inadvertent actuation of the Containment Spray System was analyzed to determine the resulting reduction in containment pressure (sudden cooling of -1.8 psid). The initial pressure condition used in this analysis was -1.7 psig. LCO 3.6.4 limits the operation of containment to equal to or less than -1.0 psig. This resulted in a minimum pressure inside containment of -2.8 psig; which is less than the design load.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment pressure satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System. An instrument uncertainty of ± 0.2 psi is conservatively included in the pressure limits to allow the use of installed instrumentation for pressure measurements.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

BASES

APPLICABILITY
(continued)

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 4 hours. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 4 hour Completion Time is reasonable to return pressure to normal.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.

REFERENCES

1. FSAR, Section 6.2.
 2. 10 CFR 50, Appendix K.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the Containment Spray and Cooling systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed to avoid exceeding peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. A spectrum of SLBs were analyzed assuming the worst single active failure. The failure to close of one Main Steam

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BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

Isolation Valve (MSIV) is the worst case single active failure for the SLB which results in the highest containment air temperature.

The limiting DBA for the maximum peak containment air temperature is an SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 120°F.

The containment design temperature is 271°F. The containment structure was analyzed to withstand the maximum peak temperature for the limiting DBA LOCA to ensure that it can contain the release of radioactive materials resulting from the accident. The containment structure was not analyzed for SLBs, which were not considered design basis for containment structural design.

The spectrum of SLBs cases are used to establish the environmental qualification operating envelope inside containment. The analysis shows that the peak containment temperature is 326°F (experienced during the MSLB at 70 % power). The performance of required safety-related equipment is evaluated against this operating envelope to ensure the equipment can perform its safety function (Ref. 2).

The temperature limit is also used in the Containment external pressure analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Containment Spray System (Ref. 1).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded. Containment average air temperature satisfies Criterion 2 of 10CFR50.36(c)(ii).

LCO

Maintaining the containment average air temperature less than or equal to the LCO temperature limit ensures that the initial containment temperature assumed in the DBA analysis will not be violated.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the

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BASES (continued)

APPLICABILITY
(continued)

probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using four temperature measurements. The four temperature measurement locations are per-selected from:

- a. TE-85 or TE-86, approximately 100 ft elevation between crane wall and containment wall,
- b. TE-87 or TE-88, approximately 100 ft elevation between steam generators,
- c. TE-89 or TE-90, approximately 140 ft elevation near equipment hatch or stairs at 270°, respectively.

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BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

- d. TE-91 or TE-92, approximately 184 ft elevation on top of steam generator missile barriers away from steam generators.

The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

- 1. FSAR, Section 6.2.
 - 2. 10 CFR 50.49.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.6 Containment Spray and Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1)

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and the Containment Cooling System provide diverse methods to limit and maintain post accident conditions to less than the containment design values.

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, containment spray is supplied by manual realignment of the residual heat removal (RHR) pumps after the RWST is empty.

The Containment Spray System provides a spray of cold borated water mixed with sodium hydroxide (NaOH) from the spray additive tank into the upper regions of containment to reduce the containment pressure and temperature, and to reduce

BASES (continued)

BACKGROUND
(continued)

fission products from the containment atmosphere during a DBA. The RWST solution temperature is an important factor in determining the heat removal capability of the Containment Spray System during the injection phase. In the recirculation mode of operation, heat is removed from the containment sump water by the RHR heat exchangers. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment atmospheric heat removal.

The Spray Additive System injects an NaOH solution into the spray. The resulting alkaline pH of the spray enhances the ability of the spray to scavenge fission products from the containment atmosphere. The NaOH added in the spray also ensures an alkaline pH for the solution recirculated in the containment sump. The alkaline pH of the containment sump water maximizes the retention of iodine and minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to the fluid.

The Containment Spray System is actuated either automatically by a containment High-High pressure signal or manually. If an "S" signal is present, the high-high pressure signal automatically starts the two containment spray pumps, opens the containment spray pump discharge valves, opens the spray additive tank outlet valves, initiates a phase "B" isolation signal, and begins the injection phase. A manual actuation of the Containment Spray System will begin the same sequence and can be initiated by operator action from the main control board. The injection phase of containment spray continues until an RWST Low-Low level alarm is received. The Low-Low level alarm for the RWST signals the operator to manually secure the system. After re-alignment of the RHR system to the containment recirculation sump, the associated RHR spray header isolation valve may be opened to allow continued spray operation of one train of spray utilizing the RHR pump to supply flow.

Containment Cooling System

Two trains of containment fan cooling, each consisting of two CFCUs with one shared CFCU for a total of five, are provided. The five CFCUs are powered from three separate vital buses, with two CFCUs on each of two vital buses and the remaining CFCU from the third vital bus. Each CFCU is supplied with cooling water from one of two separate loops of component

BASES (continued)

BACKGROUND
(continued)

cooling water (CCW). Air is drawn into the coolers through the fan and discharged to the annulus ring which supplies the steam generator compartments, pressurizer compartment, reactor coolant pumps, and outside the secondary shield in the lower areas of containment.

During normal operation, three CFCUs are operating. The fans are normally operated at high speed with CCW supplied to the cooling coils. The CFCUs are designed to limit the ambient containment air temperature during normal unit operation to less than the limit specified in LCO 3.6.5, "Containment Air Temperature." This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the DBAs.

In post accident operation following an actuation signal, the CFCUs are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere. The temperature of the CCW is an important factor in the heat removal capability of the fan units.

APPLICABLE
SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limits the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the main steam line break (MSLB). The LOCA and MSLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the worst case single failure. For the LOCA case, the worst single failure is the failure of one SSPS train, which results in only one CSP and two CFCUs available. For SLB case, the worst single failure is the failure of one MSIV to close with two CSP and three CFCUs operating.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure is 42.25 psig (experienced during an MSLB at 30% power) compared to an allowable 47 psig. The analysis shows that the peak containment temperature is 326°F (experienced during an MSLB

BASES (continued)

APPLICABLE
ANALYSIS
(continued)

at 70% power) and is compared to the environmental SAFETY qualifications of plant equipment. Both results meet the intent of the design basis. The analyses and evaluations assume a unit specific power level of 102% for the LOCA with one containment spray train and two CFCU operating. The analyses and evaluations limiting cases are based upon a unit specific power level of 30% or 70% for the MSLB with two containment spray train and three CFCUs operating for the MSLB, failure of one MSIV to close for the MSLB, and initial (pre-accident) containment conditions of 120°F and 1.3 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a -1.80 psid containment pressure decrease and is based on a sudden cooling effect of 70°F in the interior of the leak tight containment. Additional discussion is provided in the Bases for LCO 3.6.4.

The modeled Containment Spray System actuation from the containment analysis is based on a response time associated with exceeding the containment High-High pressure setpoint to achieving full flow through the containment spray nozzles. The Containment Spray System total response time includes diesel generator (DG) startup (for loss of offsite power), sequenced loading of equipment, containment spray pump startup, and spray line filling (Ref. 4).

The CFCUs performance for post accident conditions is given in Reference 4. The result of the analysis is that each train (two CFCUs) combined with one train of containment spray can provide 100% of the required peak cooling capacity during the post accident condition.

BASES (continued)

APPLICABLE
ANALYSIS
(continued)

The modeled Containment Cooling System actuation from the containment analysis is based upon a response time associated with exceeding the containment High-High pressure setpoint to achieving full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time includes signal delay, DG startup (for loss of offsite power), and component cooling water pump startup times.

The Containment Spray System and the Containment Cooling System satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

During a DBA LOCA, a minimum of two CFCUs and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits (Refs. 4). Additionally, one containment spray train is also required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two CFCU trains consisting of four CFCUs or three CFCUs each supplied by a different vital bus must be OPERABLE. Therefore, in the event of an accident, at least one train of containment spray and one train of CFCUs (two CFCUs) operate, assuming the worst case single active failure occurs. Each Containment Spray train typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an ESF actuation signal

Each CFCU includes cooling coils, dampers, fans, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and CFCUs.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System is not required to be OPERABLE in MODES 5 and 6.

BASES (continued)

ACTIONS

A.1

With one containment spray train inoperable, the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the Containment Spray System, reasonable time for repairs, and low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time for attempting restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one CFCU train inoperable such that a minimum of two CFCUs remain operable, restore the required CFCUs to OPERABLE status within 7 days. The components in this

BASES (continued)

ACTIONS
(continued)

degraded condition are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

D.1 and D.2

With one train of containment spray inoperable and one train of CFCUs inoperable such that a minimum of two CFCUs remain OPERABLE, restore one required train to OPERABLE status within 72 hours. The components remaining in OPERABLE status in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System, the iodine removal function of the Containment Spray System, and the low probability of DBA occurring during this period.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES (continued)

ACTIONS
(continued)

F.1

With two containment spray trains or one containment spray train inoperable and two CFCU trains inoperable such that one or less CFCUs remain OPERABLE, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE
REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment (only check valves are inside containment) and capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

Operating each required CFCU for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the CFCUs occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying that each required CFCU is receiving the required component cooling water flow of ≥ 1650 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 4). The component cooling water (CCW) system is hydraulically balanced during normal operation to ensure that at least 1650 gpm is delivered to each CFCU

BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

during a design bases event (DBA). The hydraulic system balance considers normal system alignments and the potential for any single active failure.

Operation of the CFCUs is permitted with lower CCW flow to the CFCUs during ASME Section XI testing or decay heat removal in MODE 4 with the residual heat removal heat exchangers in service. To support this conclusion, a calculation was performed to evaluate containment heat removal with one train of containment spray OPERABLE and reduced CCW flow to three CFCUs. The calculation concluded that this configuration would provide adequate heat removal to ensure that the maximum design pressure of containment was not exceeded during a DBA in MODE 1. This analysis also determined that a single failure could not be tolerated during this condition and still assure that the maximum design pressure of containment would not be exceeded. (Ref. 6)

The Frequency was developed considering the known reliability of the Cooling Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Part 6 of the ASME O&M Code (Ref. 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment high-high

BASES (continued)

SURVEILLANCE
REQUIREMENTS
(continued)

pressure signal with a coincident "S" signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6.7

This SR requires verification that each CFCU actuates upon receipt of an actual or simulated safety injection signal. The [18] month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

SR 3.6.6.9

The CFCUs are designed to start or restart in low speed upon receipt of an SI signal. This SR ensures that this feature is functioning properly. The 31 day frequency is selected based upon the normal operation of the CFCUs in high speed during power operation.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. 10 CFR 50, Appendix K.
3. FSAR, Section 6.2.1.

BASES (continued)

REFERENCES
(continued)

4. FSAR, Section 6.2.2.
 5. ASME, Operations and Maintenance Code, 1987 with OMa-1988 addenda, Part 6.
 6. License Amendment 89 to DPR-80 and License Amendment 88 to DPR-82, 3/2/94.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Spray Additive System

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA LOCA).

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray droplets from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Because of its stability when exposed to radiation and elevated temperature, sodium hydroxide (NaOH) is the preferred spray additive. The NaOH added to the spray also ensures a pH value of between 8.0 and 10.0 of the solution recirculated from the containment sump. This pH band minimizes the evolution of iodine as well as the occurrence of chloride and caustic stress corrosion on mechanical systems and components.

The Spray Additive System consists of one spray additive tank that is shared by the two trains of spray additive equipment. Each train of equipment provides a flow path from the spray additive tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line.

The Containment Spray System actuation signal opens the valves from the spray additive tank to the spray pump suctions or a manual containment spray initiation signal also opens the valves from the spray additive tank. The 30% to 32% NaOH by weight solution is drawn into the spray eductor suctions which inject it into the spray pump suction. The spray additive tank capacity provides for the addition of NaOH solution to all of the water sprayed from the RWST into containment. The percent solution and volume of solution sprayed into containment

(continued)

BASES

BACKGROUND
(Continued)

ensures a long term containment sump pH of ≥ 8.0 and ≤ 10.0 . This ensures the continued iodine retention effectiveness of the sump water during the recirculation phase of operation and also minimizes the occurrence of chloride induced stress corrosion cracking of the stainless steel recirculation piping.

APPLICABLE
SAFETY ANALYSES

The Spray Additive System is essential to the removal of airborne iodine within containment following a DBA LOCA.

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value volume following the accident. The analysis assumes that a minimum 83% of the containment free volume is covered by the spray (Ref. 1).

The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.6, "Containment Spray and Cooling Systems."

The DBA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable and that even so sufficient spray additive solution is added to the remaining Containment Spray System flow path to achieve the minimum required containment recirculation sump solution pH of 8.0 prior to reaching RWST low-low level.

The Spray Additive System satisfies Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA LOCA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow to raise the average long term containment sump solution pH to a level conducive to iodine retention in the liquid phase, namely, to between 8.0 and 10.0. This pH range maximizes the effectiveness of the iodine removal mechanism (from the containment atmosphere) without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In

(continued)

BASES

LCO
(continue)

addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA LOCA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the containment atmosphere iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows 48 hours for restoration of the

(continued)

BASES

ACTIONS
(continued)

Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE
REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The required volume may be surveilled using an indicated level band of 50 to 88% for the Spray Additive Tank which corresponds to the LCO 3.6.7 minimum and maximum limits adjusted conservatively for instrument accuracy of $\pm 0.3\%$. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and equipped with a low level alarm in the control room, so that there is high confidence that a level below an acceptable value would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.7.5

To ensure correct operation of the Spray Additive System, flow to the Spray Additive System eductors is verified once every 5 years by verifying that the solution flow path is not blocked from the RWST through test valve 8993 for each of the two flow paths. This SR provides assurance that NaOH will be metered into the flow path upon Containment Spray System initiation. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow rate.

REFERENCES

1. FSAR, Chapter 6.2.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.8 Hydrogen Recombiners

BASES

BACKGROUND

The function of the hydrogen recombiners is to eliminate the potential breach of containment due to a hydrogen-oxygen reaction.

Per 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammable limits would not be reached until several days after a Design Basis Accident (DBA).

Two 100% capacity independent hydrogen recombinder systems are provided. Each consists of controls located in the control room, a power supply and a recombinder. Recombination is accomplished by heating a hydrogen-air mixture above 1150°F. A single recombinder is capable of maintaining the hydrogen concentration in containment below the 4.1 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombinder is powered from a separate Engineered Safety Features bus, and is provided with a separate power panel and control panel.

APPLICABLE
SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than the lower flammable concentration of 4.1 v/o following a DBA. This control would prevent a containment wide hydrogen burn, thus ensuring the pressure and temperature assumed in the analyses are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA. Hydrogen may accumulate in containment following a LOCA as a result of:

- a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

- b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the containment sump;
- c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or
- d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 3.5 v/o about 16 days after the LOCA and 4.0 v/o about 2 days later if no recombiner was functioning (Ref. 3). Initiating the hydrogen recombiners when the primary containment hydrogen concentration reaches 3.5 v/o will maintain the hydrogen concentration in the primary containment below flammability limits.

The hydrogen recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. 4). The Hydrogen Purge System is designed and constructed such that it is Design Class I (for Quality and electrical power) but not redundant. As such, it is an adequate backup to the redundant hydrogen recombiners since it would be relied upon only in the event of a non-design basis condition.

The hydrogen recombiners satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations in these MODES. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one containment hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE hydrogen recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note that states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

(continued)

BASES (continued)

ACTIONS
(continued)

B.1 and B.2

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by the containment Hydrogen Purge System. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. In addition, the alternate hydrogen control system capability must be verified once per 12 hours thereafter to ensure its continued availability. Both the initial verification and all subsequent verifications may be performed as an administrative check by examining logs or other information to determine the availability of the key locked alternate hydrogen control system. It does not mean to perform the Surveillances are needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

C.1

If the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.6.8.1

Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to $\geq 700^{\circ}\text{F}$ in ≤ 90 minutes. After reaching 700°F , the power is

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS.
(Continued)

increased to maximum power for approximately 2 minutes and power is verified to be ≥ 60 kW.

Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.8.2

This SR ensures there are no physical problems that could affect recombinder operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failure involves loss of power, blockage of the internal flow, missile impact, etc.

A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.8.3

This SR, which is performed following the functional test of SR 3.6.8.1, requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is $\geq 10,000$ ohms.

The 18 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. 10 CFR 50.44.
 2. 10 CFR 50, Appendix A, GDC 41.
 3. FSAR Section 6.2.5.
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B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 10.3.1 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to $\leq 110\%$ of the steam generator design pressure. The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves during an overpressure event.

APPLICABLE SAFETY ANALYSES The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to $\leq 110\%$ of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 15.2 and 15.3 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO with respect to secondary system pressure. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System.

One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer relief valves and sprays. The analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The accident analysis requires that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with Reference 2.

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseal when pressure has been reduced. The OPERABILITY of the MSSVs is verified by periodic surveillance testing in accordance with the Inservice Testing Program.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, five MSSVs per steam generator are required to be OPERABLE to limit secondary pressure.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements .

Continued operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER and the Power Range Neutron Flux trip

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

setpoint so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. If one MSSV is inoperable on a SG, calculation N-114 (Ref. 8) demonstrates via RETRAN analysis, that the secondary system pressure peak resulting from the limiting AOO is < 110% of design. The transient is terminated by a reactor trip, either high pressurizer pressure or OT_ΔT, and the MSSVs maintain steam pressure below 110% at design.

When a MSSV(s) is inoperable, the power must be reduced in 4 hours (required action A.1) to a value less than or equal to the value specified in table 3.7.1-1, corresponding to the number of OPERABLE MSSVs regardless of the value of the MTC.

The Power Range Neutron Flux-high trip setpoint must also be reduced in 72 hours (Required by Action A.2), to less than or equal to the value specified in Table 3.7.1-1, corresponding to the number of OPERABLE MSSVs regardless of the value of the MTC. Required Action A.2 is modified by a Note. The Note indicates that the Power Range Neutron Flux-high trip setpoint reduction is only required in MODE 1. In MODE 1, a reduced Power Range Neutron Flux-high trip setpoint provides the required protection. In MODES 2 and 3, the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation" provide sufficient protection. Thus, reduction of the Power Range Neutron Flux-high trip setpoint is not necessary in MODE 2 or 3.

The algorithm and the RETRAN analysis used for References 7 and 9 are conservative since they both assume that the relief capacity is accordingly reduced on each SG and that the flow capacity for all inoperable MSSVs is that of the highest capacity valve.

The calculated power level is further reduced to account for instrument and channel and heat balance uncertainties and is the value specified as the MAXIMUM ALLOWABLE % RTP in Table 3.7.1-1. Per Reference 7, the calculated instrument and channel uncertainties for the power range neutron flux measurement requires a further reduction of 6% RTP to assure that the maximum RTP is not exceeded with inoperable MSSVs. Therefore, when reducing the Power Range Neutron Flux-high trip setpoint, the setpoint must be reduced to less than or equal to the % RTP value shown on Table 3.7.1-1.

The allowed Completion Times are reasonable base on operating experience to complete the Required Actions in an orderly manner without challenging unit systems.

(continued)

BASES

ACTIONS
(continued)B.1 and B.2

If THERMAL POWER or the Power Range Neutron Flux Trip is not reduced as required by Table 3.7-1 within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTSSR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 5), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 6). According to Reference 6, the following tests are required:

- a. Visual examination;
- b. Seat tightness determination;
- c. Setpoint pressure determination (lift setting);
- d. Compliance with owner's seat tightness criteria; and
- e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a + 3% setpoint (as-found lift point) tolerance on the valves for OPERABILITY (with the exception of the lowest set MSSV setpoint, which is (+3%/-2%); however, the valves are reset to $\pm 1\%$ during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

(continued)

BASES

REFERENCES

1. FSAR, Section 10.3.1.
 2. Not Used.
 3. FSAR, Section 15.2 and 15.3.
 4. NRC Information Notice IN-94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.
 5. ASME, Boiler and Pressure Vessel Code, Section XI.
 6. ANSI/ASME OM-1-1987, (including OM-a-1988 ADDENDA).
 7. Westinghouse Report WCAP-11082, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Stations Eagle 21 Version", dated May 1993.
 8. PG&E Design Calculation N-114, "Over-Pressure Study for One MSSV Per Loop Unavailable", dated 3/10/94.
 9. PG&E Design Calculation N-115, "Reduced Power Levels for A Number of MSSVs Inoperable", dated 3/14/94.
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B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are installed back to back with the MS reverse flow check valves. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by high negative steam line pressure rate or low steam line pressure or high-high containment pressure. The MSIVs are held in the open position and will fail in the closed direction on loss of control air and fail open on loss of actuation power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section 10.3 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the FSAR, Section 6, Appendix 6.2 C (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the FSAR, Section 15.4.2 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for the containment pressure analysis is the SLB inside containment, with initial reactor power at 30% with no loss of offsite power, and failure of the MSIV on the affected steam generator to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

containment. Due to the assumed reverse flow (the MSIV reverse flow check valves are not credited to function even though they are Design Class I) and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIV contribute to the total release. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available, and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an MSIV to close.

The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:

- a. An HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is assumed to be discharged into containment from all steam generators, as no credit is taken for the MSIV reverse flow check valves, until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.
- b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.
- c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.
- d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that four MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits or the NRC staff approved licensing basis.

APPLICABILITY

The MSIVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed and de-activated (vented or prevented from opening), when there is significant mass and energy in the RCS and steam generators. When the MSIVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low, thus OPERABILITY in MODE 4 is not required.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the MSIV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

(continued)

BASES

ACTIONS
(continued)B.1

If the MSIV cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Times are reasonable, based on operating experience, to reach MODE 2 and to close the MSIVs in an orderly manner and without challenging unit systems.

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis. MSIV closure is indicated by the control room valve indicating lights or monitor light box lights.

C.1 and C.2

The 8 hour Completion Time is consistent with that allowed in Condition A.

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, the inoperable MSIVs must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

D.1 and D.2

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.7.2.1

This SR verifies that MSIV closure time is \leq 5.0 seconds on an actual or simulated actuation signal. The remote manual hand switch may be used as the actuation signal for this SR. The MSIV closure time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1 or 2.

SR 3.7.2.1

The Frequency is in accordance with the Inservice Testing Program.

This test may be conducted in MODE 3 with the unit at operating temperature and pressure: This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. However, the test is normally conducted in MODE 5 as permitted by the cold shutdown frequency justification provided in the Inservice Testing Program (IST) and as permitted by Reference 6, Part 10, paragraph 4.2.1.2(c).

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The frequency of MSIV testing is every 18 months. The 18 month Frequency is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Section 6, Appendix 6.2 C.
3. FSAR, Section 15.4.2.
4. 10 CFR 100.11.
5. ASME, Boiler and Pressure Vessel Code, Section XI.
6. ANSI/ASME OM-1-1987, (including OM-a-1988 ADDENDA).

B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation Valves (MFIVs) and Main Feedwater Regulation Valves (MFRVs) and Associated Bypass Valves

BASES

BACKGROUND

The safety related function of the MFRVs and the associated bypass valves is to provide the initial isolation of main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break (HELB). Since the MFRVs and associated bypass valves are located in non-safety related piping, the MFIVs also provide safety related isolation of the MFW flow to the secondary side of the steam generators a short time later. Closure of the MFIVs or MFRVs and associated bypass valves terminates flow to the steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs or MFRVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs or MFRVs and associated bypass valves, effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment, and reducing the cooldown effects for SLBs.

The MFIVs isolate the non-safety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact loops.

One MFIV and one MFRV and its associated bypass valve, are located on each MFW line, outside but close to containment. The MFIVs and MFRVs are located upstream of the AFW injection point so that AFW may be supplied to the steam generators following MFIV or MFRV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases, and refilled prior to AFW reaching the steam generator following either an SLB or FWLB.

The MFIVs and MFRVs and associated bypass valves, close on receipt of any safety injection (SI) signal, or steam generator (S/G) water level-high high signal. They may also be actuated manually. The Main Feedwater Pump Turbine is also tripped upon receipt of an SI or S/G water level-high high signal (as well as other pump related trips), however, these are Class II trips and are only credited as a backup to the single failure of a MFRV and associated bypass valve trip. The MFRVs and associated bypass valves also close on receipt of a T_{avg} -Low coincident with reactor trip (P-4). In addition to the MFIVs and the MFRVs and associated bypass valves, a check valve located upstream of the MFIV is available. The check valve isolates the feedwater line, penetrating containment, and ensures that the intact steam

(continued)

BASES

BACKGROUND
(continued)

generators do not continue to feed the feedwater line break in the non-safety related piping upstream of the feedwater isolation check valves and that the AFW flow will be to the steam generators.

A description of the MFIVs and MFRVs is found in the FSAR, Section 10.4.7 (Ref. 1).

APPLICABLE
SAFETY ANALYSES

The design basis of the MFIVs and MFRVs is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs or MFRVs and associated bypass valves, is relied on to terminate an SLB for core and containment response analysis and excess feedwater event upon the receipt of a feedwater isolation signal on high-high steam generator level.

Failure of an MFIV, MFRV, or the associated bypass valves to close following an SLB or FWLB can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs and MFRVs satisfy Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

This LCO ensures that the MFIVs, MFRVs and their associated bypass valves will isolate MFW flow to the steam generators, following an FWLB or main steam line break. The MFIVs will also isolate the non-safety related portions from the safety related portions of the system.

This LCO requires that four MFIVs and four MFRVs and associated bypass valves be OPERABLE. The MFIVs and MFRVs and their associated bypass valves are considered OPERABLE when isolation times are within limits and they close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. A feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event and failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY

The MFIVs and MFRVs and the associated bypass valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and steam generators. This ensures that, in the event of an HELB, a single failure cannot result in the

(continued)

BASES

APPLICABILITY
(continued)

blowdown of more than one steam generator. In MODES 1, 2, and 3, the MFIVs and MFRVs and the associated bypass valves are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve, they are already performing their safety function.

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the MFIVs, MFRVs, and the associated bypass valves are normally closed since MFW is not required.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFIV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFIVs that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

B.1 and B.2

With one MFRV in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable MFRVs, that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls to ensure that the valves are closed or isolated.

C.1 and C.2

With one associated bypass valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the Class II main feedwater pump trip and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 72 hour Completion Time is reasonable, based on operating experience.

Inoperable associated bypass valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

D.1

With two inoperable valves in the same flow path, only the Class II main feedwater pump trip is available to operate automatically and perform the required safety function. Although the containment can be isolated with the failure of two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path, and as such, is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the condition where at least one valve in each flow path is performing the required safety function. The 8 hour Completion Time is reasonable, based on operating experience, to complete the actions required to close the MFIV or MFRV, or otherwise isolate the affected flow path.

(continued)

BASES

ACTIONS
(continued)

E.1 and E.2

If the MFIV(s) and MFRV(s) and the associated bypass valve(s) cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.3.1 and SR 3.7.3.2

These SRs verify that the closure time of each MFIV is ≤ 60 seconds and that each MFRV, and associated bypass valves is ≤ 7 seconds. The MFIV closure times are assumed in the accident and containment analyses. These Surveillances are normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. This is consistent with the ASME Code (Ref. 2) stroke requirements during operation in MODES 1 and 2.

The Frequency for these SRs is in accordance with the Inservice Testing Program.

SR 3.7.3.3

This SR verifies that each MFIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The Frequency of MFIV testing is every 18 months. The 18 month Frequency is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.4.7.
 2. ANSI/ASME OM-1-1987, (including OM-a-1988 ADDENDA).
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B 3.7 PLANT SYSTEMS

B 3.7.4 10% Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The 10% ADVs (PCV-19, PCV-20, PCV-21 and PCV-22) provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Bypass System to the condenser not be available, as discussed in the FSAR, Section 15 (Ref 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST) and firewater storage tank (FWST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Dump System.

One ADV line for each of the four steam generators is provided. Each ADV line consists of one ADV and an associated manual block valve.

The ADVs are provided with upstream manual block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are normally provided with a non-Class I pressurized supply of air. With a loss of pressure in the normal air supply the backup non-Class I nitrogen supply, automatically supplies to operate the ADVs. With the loss of both the normal air supply and the backup nitrogen supply, the normal supplies are blocked and the Class I backup air bottle system is activated. With the backup air bottle system activated, control of the valves is remote manual via the Class I control circuit from the Control Room. The nitrogen bottled air supply is sized to provide sufficient pressurized gas to operate the ADVs for the time required for Reactor Coolant System cooldown to RHR entry conditions. In addition, handwheels are provided for local manual operation.

APPLICABLE
SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions at the maximum allowable rate of 100°F per hour. The ADVs support the AFW cooldown function from normal zero-load temperature in the RCS to a hot-leg temperature of 350°F (which is the maximum temperature allowed for placing the RHR system in service). Various cooldown rates are applicable depending upon the event and the assumed available equipment. These rates vary from a high of 100°F/hr for the SGTR event to 25°F/hr for a natural

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

circulation cooldown event utilizing the cooling water supply available in the CST and FWST.

The ADVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for events accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the ADVs and main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR and thus limit offsite dose, is more critical than the time required to cool down to RHR conditions for this event and also for other events. Thus, the SGTR is the limiting event for the ADVs. All four ADVs are required to be OPERABLE to satisfy the SGTR accident analysis requirements since the SGTR event assumes that the ADV on the faulted SG fails open to maximize the offsite dose and that the three intact SGs are utilized to cool the RCS at the Maximum allowable rate of 100°F/hr.

The once per 24 hour verification that backup air bottle pressure is greater than or equal to 260 psig assures that the ADVs will perform as required by the applicable safety analyses.

The ADVs are equipped with manual block valves in the event an ADV spuriously fails open or fails to close during use.

The ADVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Four ADV lines are required to be OPERABLE. One ADV line is required from each of four steam generators to ensure that at least two ADV lines are available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of an ADV on an unaffected steam generator. The block valves must be OPERABLE to isolate the failed open ADV line. A closed block valve renders its ADV line inoperable, and the appropriate ACTION must be entered until such time that the block valve is opened.

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

(continued)

BASES

LCO
(continued)

An ADV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand:

APPLICABILITY

In MODES 1, 2, and 3, all four ADVs are required to be OPERABLE. In MODE 4, only the ADVs associated with the steam generators being relied upon for heat removal, are required to be OPERABLE.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one required ADV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ADV lines, a non-safety grade backup in the Steam Bypass System, and MSSVs and is based on a PRA analysis and the low probability of a SGTR and LOOP event occurring during this period that would require the ADV lines. Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

B.1

With two ADV lines inoperable, action must be taken to restore all but one ADV line to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 72 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines.

C.1

With three or more ADV lines inoperable, action must be taken to restore all but two ADV lines to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines.

(continued)

BASES

ACTIONS
(continued)

D.1 and D.2

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

Plant procedures provide a 31 day verification that the 10% ADV manual block valves are open assures that the valves have not been inadvertently closed.

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened and closed remotely using the remote manual controls and the backup air bottles. This SR ensures that the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components are expected to pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.2

While not a safety function, the function of the manual block valve is to isolate a failed open ADV or isolate an ADV for repair or testing during plant operation. Cycling the block valve both closed and open demonstrates its capability to perform this function. Operating experience has shown that these components are expected to pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.3

The function of the back-up air bottles is to assure that the ADVs will be able to be opened as required to perform a controlled cooldown of the RCS in the event of a loss of the normal air supply system. The backup air bottle system was specifically installed to allow the RCS to be cooled for a SGTR coincident with a loss of offsite power. Verification of the bottle pressure once every 24 hours allows for timely bottle replacement and trending for leaks.

(continued)

BASES

REFERENCES

1. FSAR, Section 15.
 2. WCAP-11723
 3. DCM S-25B, S-3B, AND S-4.
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B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps take normal suction through valve MU-671 on the single suction line from the condensate storage tank (CST) (LCO 3.7.6) (this valve must remain open for the applicable accident analysis assumptions to be valid) and are capable of being aligned to the firewater storage tank (FWST) (LCO 3.7.6) and pump to the steam generator secondary side via separate and independent connections to the main feedwater (MFW) piping outside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the condenser steam dump valves and recirculated to the CST.

The AFW System consists of two motor driven AFW pumps and one steam turbine driven pump configured into three trains. Each motor driven pump provides 100% of AFW flow capacity, and the turbine driven pump provides 200% of the required capacity to the steam generators, with 100% capacity defined as the flow required to two steam generators during the AFW design basis accident analysis (loss of normal feedwater flow (Ref. 1)). The pumps are equipped with recirculation lines to prevent pump operation against a closed system. Each motor driven AFW pump is powered from an independent Class 1E power supply and feeds two steam generators, although each pump has the capability to be manually realigned to feed other steam generators. The steam turbine driven AFW pump receives steam from two main steam lines upstream of the main steam isolation valves. Each of the steam feed lines will supply 100% of the requirements of the turbine driven AFW pump.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

The turbine driven AFW pump supplies a common header capable of feeding all steam generators with vital AC powered control valves. One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator

(continued)

BASES

BACKGROUND
(continued)

pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

The AFW System (both the one turbine-driven and two motor-driven AFW pumps) actuates automatically upon actuation of the anticipated transient without scram mitigating system actuation circuitry (AMSAC). The motor-driven pumps are additionally actuated by: (1)safety injection; (2) an associated bus transfer to the diesel generator signal; (3)a trip of both MFW pumps; or (4) steam generator water level - low-low in one of four SGs. The turbine-driven pump is additionally actuated by 12 kV bus undervoltage or steam generator low-low level in two of four SGs via ESFAS (LCO 3.3.2).

The AFW System is discussed in the FSAR, Section 6.5 (Ref. 1).

APPLICABLE
SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater:

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to at least two steam generators at pressures corresponding to the lowest steam generator safety valve set pressure plus 3% tolerance plus 3% accumulation within 1 minute after event initiation.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions. Sufficient AFW flow must also be available to account for flow losses such as pump recirculation and AFW spillage through feedwater line breaks.

The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows:

- a. Feedwater or Main Steam Line Break (FWLB); and
- b. Loss of MFW (the coincident loss of offsite power is a less limiting transient since RCP heat input is lost).

In addition, the minimum available AFW flow and system characteristics must be considered in the analysis of normal cooldown and small break loss of coolant accident (LOCA) due to their potential impact.

The AFW System is also designed for decay heat removal following a Steam Generator Tube Rupture (SGTR). As such the steam turbine driven AFW pump has redundant steam supplies to assure continued availability following a SGTR or MSLB event.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment on loss of MFW, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. One motor driven AFW pump would deliver to the broken MFW header at the pump maximum flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine driven pump when required to ensure an adequate feedwater supply to the steam generators during loss of power. Vital AC power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. Three independent AFW pumps in three diverse trains are required to be OPERABLE to ensure the availability of decay and residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses and having the third AFW pump powered by a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs. To assure steam turbine driven AFW pump operability via redundant steam supplies, steam traps 104, 105 and 106 on the supply lines must be operable or bypassed to ensure adequate condensate removal and check valves MS-6166 and MS-6167 must be operable.

The AFW System supply is configured into three trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps, each powered by a separated vital bus, be OPERABLE in two diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of two main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

The operability of the AFW suction flow path is assured by verifying the condensate storage tank outlet valve open and by

(continued)

BASES

LCO
(continued)

verifying the capability to align the fire water storage tank to the AFW pump suction.

The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

ACTIONS

A.1

If one of the two steam supplies to the turbine driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

- a. The redundant OPERABLE steam supply to the turbine driven AFW pump;
- b. The availability of redundant OPERABLE motor driven AFW pumps; and
- c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

If Condition A, an inoperable steam supply to the turbine driven AFW pump, is entered while, for instance, motor driven AFW pump 1-2 is inoperable and the motor driven AFW pump 1-2 is subsequently returned to an OPERABLE condition shortly after Condition A is entered, the LCO may already have not been met for up to 72 hours. This could lead to a total of up to 10 days for

(continued)

BASES

ACTIONS
(continued)

restoration of the motor driven AFW pump 1-2 and the turbine driven AFW pump steam supply. If before the steam supply is returned OPERABLE motor driven AFW pump 1-3 becomes inoperable, the AFW system could be inoperable for as long as 13 days.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

With one of the required AFW trains (pump or flow path) inoperable in MODE 1, 2, or 3 for reasons other than Condition A, action must be taken to restore OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to the turbine driven AFW pump. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1 and C.2

When Required Action A.1 or B.1 cannot be completed within the required Completion Time, or if two AFW trains are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

In MODE 4 with two AFW trains inoperable, operation is allowed to continue because only one motor driven pump AFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate RHR.

(continued)

BASES

ACTIONS
(continued)

D.1

If all three AFW trains are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with non-safety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

E.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops - MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

F.1

With the CST supply to the AFW pump suction unavailable, the primary safety related source of water required for cooldown is unavailable. The four hours required to restore the CST supply to the AFW pumps is a reasonable time to limit the risk from an event requiring the plant to cooldown.

G.1

With the FWST supply incapable of alignment to the AFW suction the required additional source of water needed for a natural circulation cooldown is unavailable. The seven days required to restore the ability for FWST realignment is a reasonable time to limit the risk from an event requiring a natural circulation cooldown based upon the available non-Class I AFW sources.

H.1 and H.2

When the Required Action F.1 or G.1 cannot be completed within the Required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in

(continued)

BASES (continued)

ACTIONS

H.1 and H.2 (continued)

MODE 4 without reliance upon the steam generator for heat removal within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency, based on engineering judgment, is consistent with procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code (Ref. 2) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that the SR for the turbine-driven AFW pump should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states the SR is not required in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the required AFW train may already be aligned and operating.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR for the turbine-driven pump can be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that the SR is not required in MODE 4. In MODE 4, the required motor-driven pump is already operating and the autostart function is not required. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.

(continued)

BASES

SUREVILLANCE
REQUIREMENTS
(continued)

SR 3.7.5.5
Not Used.

SR 3.7.5.6

This SR verifies that the FWST is capable of being aligned to the AFW pump suction. This assures that this additional supply of required AFW is available from the seismically qualified FWST should it be needed for a natural circulation cooldown.

The 92 day frequency, based on engineering judgement, is consistent with procedural controls governing valve operation, and ensures correct valve positions.

A similar SR is not required for the CST alignment since the AFW system is used for startup and an AFW pump is tested each month. This operation and the pump tests assure proper valve alignment.

REFERENCES

1. FSAR, Section 6.5 and section 15.2.8.
 2. ANSI/ASME OM-1-1987, (including OM-a-1988 ADDENDA).
 3. DCM S-3B
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B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tank (CST) and Fire Water Storage Tank (FWST)

BASES

BACKGROUND

The CST supplemented by the FWST provide a safety grade source of water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST and FWST provide a passive flow of water, by gravity, to the Auxiliary Feedwater (AFW) System (LCO 3.7.5). The steam produced is released to the atmosphere by the main steam safety valves or the atmospheric dump valves if the main steam isolation valves are closed. The AFW pumps operate with a continuous recirculation to the CST.

When the main steam isolation valves are open, the preferred means of heat removal is to discharge steam to the condenser by the nonsafety grade path of the condenser dump valves. The condensed steam is returned to the CST by the condensate pumps. This has the advantage of conserving condensate while minimizing releases to the environment.

Because the CST and FWST are the principal components for removing residual heat from the RCS, they are designed to withstand earthquakes and other natural phenomena, including missiles that might be generated by natural phenomena. The CST and FWST are designed to Seismic Category I to ensure availability of the feedwater supply. Feedwater is also available from alternate sources as described in the FSAR.

A description of the CST is found in the FSAR, Section 9.2.6 (Ref. 1).

APPLICABLE
SAFETY ANALYSES

The CST and FWST provide cooling water to remove decay heat and to cool down the unit following all events in the accident analysis as discussed in the FSAR, Chapters 6 and 15 (Refs. 2 and 3, respectively). The limiting event for AFW supply, i.e., CST and FWST minimum volumes, is based on a loss of offsite power which assumes a reduced Reactor Coolant System (RCS) cooldown rate and requires seismically qualified water sources. The lower RCS cooldown rate on natural circulation increases the cooldown period until the residual heat removal (RHR) system can be used to remove further decay heat. The extended cooldown time thus requires more AFW supply than can be provided by the seismically qualified portion of the CST.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

Other events requiring condensate volume are:

- 1) the large feedwater line break coincident with a loss of offsite power. Single failures that also affect this event include the following:
 - a. Failure of the diesel generator powering the motor driven AFW pump to the unaffected steam generator (requiring additional steam to drive the remaining AFW pump turbine); and
 - b. Failure of the steam driven AFW pump (requiring a longer time for cooldown using only one motor driven AFW pump).

These are not usually the limiting failures in terms of consequences for these events.

and,

- 2) a break in either the main feedwater or AFW line near where the two join. This break has the potential for dumping condensate until terminated by operator action, since the Emergency Feedwater Actuation System would not detect a difference in pressure between the steam generators for this break location. This loss of condensate inventory is partially compensated for by the retention of steam generator inventory.

The CST satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

To satisfy Hosgri analysis assumptions, the CST and FWST must contain sufficient cooling water to remove decay heat following a reactor trip from 102% RTP, and then to cool down the RCS to RHR entry conditions, assuming a coincident loss of offsite power and the most adverse single failure. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine.

The CST level required is equivalent to a usable volume of \geq 41.3% indicated level (164,678 gallons). The FWST level required is equivalent to a usable volume of \geq 41.75% indicated level (115,844 gallons) for two units operating and \geq 22.2% indicated level (57,922 gallons) for one unit operating. These levels are based on holding the unit in MODE 3 for 2 hours, followed by a natural circulation cooldown to RHR entry conditions at 25°F/hour. This basis is established in Reference 4.

The OPERABILITY of the CST and FWST is determined by maintaining the tank levels at or above the minimum required levels.

(continued)

BASES (continued)

APPLICABILITY In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the CST and FWST are required to be OPERABLE.

In MODE 5 or 6, the CST or FWST are not required because the AFW System is not required.

ACTIONS A.1 and A.2

If the CST or FWST levels are not within limits, the CST level must be restored within 4 hours and the FWST level must be restored within 7 days to return the AFW sources to an OPERABLE status. The CST must be restored to OPERABLE status within 4 hours because it is the primary Class 1 AFW supply. The 4 hour Completion Time provides time to restore the required CST level from the condenser or other source, and is a reasonable time to limit the risk from accidents or events requiring the plant to cooldown. The 7 day Completion Time provides time to restore the required FWST level is a reasonable time to limit the risk of a natural circulation cooldown event that would require the use of the backup volume in addition to the volume contained in the CST. Alternate non-seismically qualified water sources are also available to supply water to supplement the CST volume.

B.1 and B.2

If the CST or FWST cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on the steam generator for heat removal, within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

**SURVEILLANCE
REQUIREMENTS**SR 3.7.6.1

This SR verifies that the CST contains the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST levels.

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BASES (continued)

SURVEILLANCE
REQUIREMENTS
(Continued)

SR 3.7.6.2

This SR verifies that the FWST contain the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the FWST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the FWST levels.

REFERENCES

1. FSAR, Section 9.2.6 and 9.5.1.
 2. FSAR, Chapter 6.
 3. FSAR, Chapter 15.
 4. DCM S-3B
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B 3.7 PLANT SYSTEMS

B 3.7.7 Vital Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel storage pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Auxiliary Saltwater (ASW) System, and thus to the environment.

The CCW system consists of three CCW pumps powered by separate vital buses, two CCW heat exchangers, and an internally baffled two chamber CCW surge tank. The piping system consists of three parallel loops headered out of the two parallel 100% capacity heat exchangers. Two loops are separable redundant full capacity vital service loops which serve only ESF equipment and the non-redundant post-LOCA sample cooler. The vital loops are separable to mitigate a single passive failure during post LOCA long term cooling. A third miscellaneous service loop serves nonvital equipment. The divided surge tank is connected to the vital header return piping and is sized to meet system leakage requirements and maintain adequate NPSH on system pumps.

The CCW system is hydraulically balanced to ensure that sufficient cooling water is delivered to ESF loads on the vital loops during a DBA.

The CCW system is designed to perform its function with a single failure of any component. All three pumps are automatically started on receipt of a safety injection signal, and the miscellaneous service loop is automatically isolated on hi-hi containment pressure.

Additional information on the design and operation of the system, along with a list of the components served, is presented in the FSAR, Section 9.2.2 (Ref. 1). The principal safety related function of the CCW System is the removal of accident generated containment heat via the containment fan cooling units (CFCUs) and removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. Decay heat removal may be during a normal or post accident cooldown and shutdown.

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one CCW loop to remove the post DBA heat load from the containment, without exceeding the design basis continuous CCW temperature of

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

120°F with an allowable transient not to exceed 140°F for more than 6 hours (Ref. 1).

In accordance with GDC 44, the CCW system is designed to provide sufficient heat removal for normal and post accident ESF heat loads without overheating. The CCW system and ASW system are essentially considered a single heat removal system for the purpose of assessing the ability to sustain either a single active or passive failure and still perform design basis heat removal. Only one ASW pump and one CCW heat exchanger is required, as assumed in the safety analysis, to provide sufficient heat removal from containment to mitigate a DBA. However, to ensure maximum heat removal capability, operators are instructed to place the second CCW heat exchanger in service early in the emergency operating procedures.

The CCW System also functions to cool the unit from RHR entry conditions ($T_{ave} < 350^{\circ}\text{F}$), to MODE 5 ($T_{ave} < 200^{\circ}\text{F}$), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW heat exchangers and RHR heat exchangers operating.

In the event that CCW system leakage occurs and system makeup is not available, the surge tank volume provides a minimum of 20 minutes, based on a non-mechanistic leakage rate of 200 gpm, for operators to locate and isolate the leak or realign the CCW system into two separate loops before the system becomes impaired due to water loss.

The CCW System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

In the event of a DBA, one CCW loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two vital loops of CCW must be OPERABLE. At least one CCW loop will operate assuming the worst case single active failure occurs coincident with a loss of offsite power. To meet the LCO on Component Cooling Water loops, vital headers A and B, both CCW heat exchangers, the surge tank, and all three CCW pumps must be operable.

A vital CCW loop is considered OPERABLE when:

- a. Two CCW pumps, one CCW heat exchanger, and the surge tank are OPERABLE; and
- b. The associated piping, valves, and instrumentation and controls required to perform the safety related function are OPERABLE.

(continued)

BASES

LCO
(continued)

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System, except for isolation of CCW to the CFCUs. Isolation of CCW to the CFCUs could potentially affect the flow balance and requires evaluation to ensure continued operability.

Split loop alignment of the CCW system during normal operation requires Condition A to be entered because the CCW system cannot tolerate a single failure in this configuration.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its principal safety related function of removal of accident generated containment heat via the CFCUs and removal of decay heat from the reactor via the Residual Heat Removal (RHR) System.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable vital CCW loop results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one vital CCW loop is inoperable, action must be taken to restore two vital CCW loops to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE vital CCW loop is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the overall heat transfer capability of ultimate heat sink system, operator action, and the low probability of a DBA occurring during this period.

Split loop alignment of the CCW system during normal operation requires Condition A to be entered because the CCW system cannot tolerate a single failure in this configuration.

B.1 and B.2

If the vital CCW loop cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required

(continued)

BASES (continued)

ACTIONS

B.1 and B.2 (continued)

unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW System. A possible exception to this note, is isolation of CCW to the CFCUs. Isolation of CCW to the CFCUs could potentially affect the flow balance and requires evaluation to ensure continued operability.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

(continued)

BASES

SUREVILLANCE
REQUIREMENTS
(continued)

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated safety related actuation signal. The 18 month Frequency is based on the need to perform this surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This surveillance requirement applies to the SIS auto-start and the 4kV auto-transfer automatic starts only. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.2.2.
 2. FSAR, Section 6.2.
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B 3.7 PLANT SYSTEMS

B 3.7.8 Auxiliary Saltwater System (ASW)

BASES

BACKGROUND

The ASW provides a heat sink from the Pacific Ocean for the removal of process and operating heat from safety related components via the component cooling water (CCW) system during all modes of operation including a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the ASW also provides this function for various non-safety related components. The safety related function is covered by this LCO.

The ASW consists of two, 100% capacity, safety related, cooling water trains. Each train consists of one 100% capacity pump, one component cooling water (CCW) heat exchanger, piping, valving, and instrumentation. The pumps are automatically started upon receipt of a safety injection signal or 4kV automatic transfer. Normal configuration is for one train operation with the second pump cross-tied in stand-by and the second heat exchanger valved out-of-service except when the UHS temperature is 64°F or higher; therefore no valve realignment occurs with a safety injection signal. Manual and remote manual system realignment provides for utilization of the second CCW heat exchanger, for use of the standby pump on the same unit, for cross-tying the standby ASW pump from opposite unit, and for train separation for long term cooling. The ASW unit cross-tie valve (FCV-601) allows one ASW pump on one unit to supply the CCW heat exchanger(s) on the other unit. FCV-601 is controlled by ECG 17.1.

Additional information about the design and operation of the ASW system, is presented in the FSAR, Section 9.2.7 (Ref. 1). The principal safety related function of the ASW system is the removal of decay heat from the reactor via the vital CCW System.

APPLICABLE
SAFETY ANALYSES

The design basis of the ASW system is for one ASW train, in conjunction with the CCW System and the containment cooling systems, to remove accident generated and core decay heat following a design basis LOCA as discussed in the FSAR, Section 6.2 (Ref. 2). The ASW system can be re-configured to maintain the CCW temperature to within its design bases limits. The ASW system is designed to perform its function with a single failure of any active component, with or without the loss of offsite power. This assumes a maximum ASW temperature of 64°F occurring simultaneously with maximum heat loads on the system.

The ASW system, in conjunction with the CCW System, also cools the unit from residual heat removal (RHR) entry conditions to MODE 5 during normal and post accident operations. The time required for this evolution is a function of the number of ASW

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

pumps, CCW heat exchangers, and RHR heat exchangers that are operating.

The ASW system satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two ASW trains are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming that the worst case single active failure occurs coincident with the loss of offsite power.

An ASW train is considered OPERABLE during MODES 1, 2, 3, and 4 when:

- a. The pump is OPERABLE; and
 - b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are capable of performing their intended safety functions. The standby crosstied ASW pump discharge (via FCV 495 and 496) provides redundancy for the operating ASW pump. The standby heat exchanger, valved out for operating convenience, is available to provide additional heat removal capability by valving in the second heat exchanger. Valving is provided to permit train separation during the long term cooling phase of a LOCA should a single passive failure occur.
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APPLICABILITY

In MODES 1, 2, 3, and 4, the ASW system is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the ASW system and required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ASW system are determined by the systems it supports.

ACTIONS

A.1

If one ASW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE ASW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE ASW train could result in loss of ASW system function. The Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," should be entered if an inoperable ASW train results in an inoperable decay heat removal train. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

(continued)

BASES

ACTIONS
(continued)B.1 and B.2

If the ASW train cannot be restored to OPERABLE status within the associated Completion Times, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTSSR 3.7.8.1

Verifying the correct alignment for manual and power operated valves in the ASW system flow path provides assurance that the proper flow paths exist for ASW system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position and that those valves requiring remote positioning have available power and air supplies such that if required, the valve would be capable of being placed in its required position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.8.2

This SR verifies proper remote manual full stroke operation of the ASW valves. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 92 day Frequency is based on the IST program frequency and is consistent with the ASME O&M Code testing requirements, and ensures the ability to correctly align the valves. Operating experience has shown that these components usually pass the Surveillance when performed at

(continued)

BASES

**SUREVILLANCE
REQUIREMENTS**SR 3.7.8.2 (continued)

the 92 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

This SR verifies proper automatic operation of the ASW pumps on an actual or simulated safety related actuation signal. The ASW is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This surveillance requirement applies to the SIS auto-start and the 4kV auto transfer automatic starts only. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.2.7.
 2. FSAR, Section 6.2.
 3. NRC GENERIC LETTER 91-13, "REQUEST FOR INFORMATION RELATED TO THE RESOLUTION OF GENERIC ISSUE 130, 'ESSENTIAL SERVICE WATER SYSTEM FAILURES AT MULTI-UNIT SITES,' PURSUANT TO 10 CFR 50.54 (f)," dated September 19, 1991.
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B 3.7 PLANT SYSTEMS

B 3.7.9 Ultimate Heat Sink (UHS)

BASES

BACKGROUND

The UHS provides a heat sink for transferring heat from safety related components during a transient or accident, as well as safety related and non-safety related heat loads during normal operation. This is done by utilizing the Pacific Ocean, the Auxiliary Saltwater System (ASW) and the Component Cooling Water (CCW) System.

The UHS is common to both units and has been defined as the Pacific Ocean. The principal functions of the UHS are dissipation of heat during normal operation, dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

The basic performance requirements are that a 30 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. To ensure UHS availability, ASW components located within the projected sea wave zone are designed to operate during extreme ocean levels for a short duration (for example, tsunami run up and draw down conditions) per Reference 2. To maintain adequate cooling for safety related equipment, operational limits are established based on ocean supply temperature per Reference 4.

Additional information on the design and operation of the system along with a list of components served, can be found in Reference 1.

APPLICABLE
SAFETY ANALYSES

The UHS is the sink for heat removed from the reactor core and containment following all accidents and anticipated operational occurrences in which the unit is cooled down and placed on residual heat removal (RHR) operation. The Pacific Ocean as a single water source for the Ultimate Heat Sink will provide in excess of 30 days of cooling water during normal and emergency shutdown conditions as required by AEC Safety Guide 27 (Ref. 3).

The UHS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The UHS is required to be OPERABLE and is considered OPERABLE if it is at or below the maximum temperature that would allow the ASW to operate for at least 30 days following the DBA without exceeding the maximum design temperature of the CCW system. To meet this condition, the UHS temperature should not exceed 64°F unless two CCW heat exchangers are in service during normal unit operation.

(continued)

BASES (continued)

APPLICABILITY In MODES 1, 2, 3, and 4, the UHS is required to support the OPERABILITY of the equipment serviced by the UHS and required to be OPERABLE in these MODES.

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1

If the UHS is inoperable (i.e., inlet water temperature > 64°F), action of placing a second CCW heat exchanger in service must be performed within 8 hours. This action provides assurance that the ASW system and the CCW system can operate within its temperature limit.

The 8 hour Completion Time is reasonable based on the low probability of an accident occurring during the 8 hours that the temperature is > 64°F without two CCW heat exchangers in service and the time required to reasonably complete the Required Action.

B.1 and B.2

If the the second heat exchanger cannot be placed in service within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.9.1
Not Used.

SR 3.7.9.2

This SR verifies that adequate long term (30 day) cooling can be maintained. The 24, 12 and 2 hour surveillance Frequencies are based on operating experience related to trending of the temperature variations during the applicable MODES. This SR verifies the temperature of the UHS so that appropriate actions can be taken to assure that the ASW system can continue to assure that the CCW system will not exceed its design temperature profile.

(continued)

BASES (continued)

- REFERENCES
1. FSAR, Section 9.2.5.
 2. FSAR, Sections 2.4.11.5 & 2.4.11.6.
 3. AEC Safety Guide 27.
 4. SSER 16.
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B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Ventilation System (CRVS)

BASES

BACKGROUND

The CRVS provides a protected environment from which operators can control the units from the common control room following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The CRVS consists of two independent, redundant trains that recirculate and filter the control room air (one train from each unit). Each train consists of a heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and one pressurization supply fan, one filter booster fan, and one main supply fan. Ductwork, dampers, and instrumentation also form part of the system.

The CRVS is an emergency system, parts of which may also operate during normal unit operations. Upon receipt of an actuating signal, the normal air supply to the control room is isolated, and the stream of outside ventilation air from the pressurization system and recirculated control room air is passed through the system filter. The pressurization system draws outside air from either the north end or the south end of the turbine building based upon the wind direction or the absence of releases at the inlet. The prefilters remove any large particles in the air, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each filter train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. The heater is important to the effectiveness of the charcoal adsorbers.

Manual or automatic actuation of the CRVS places the system in one of three states; 1) pressurization (MODE 4), 2) recirculation (MODE 3), or 3) smoke removal (MODE 2).

Actuation of the system to the recirculation mode closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the control room air through the redundant trains of HEPA and the charcoal filters. The pressurization mode also initiates pressurization and filtered ventilation of the air supply to the control room.

Outside air is filtered, diluted via pressure equalization with air from the mechanical equipment room, and added to the air being recirculated from the control room. Pressurization of the control room prevents infiltration of unfiltered air from the surrounding areas. The actions taken in the manual actuation of the recirculation mode are the same, except that the signal switches control room ventilation to an isolation alignment to prevent outside air from entering the control room.

(continued)

BASES

BACKGROUND
(continued)

To monitor the status of the booster fan(s) small plastic streamers are installed on the exhaust duct grates. These exhaust ducts are located in the back of the control room in the ceiling and are used to take suction on the control room atmosphere. These streamers will hang down when the booster fan(s) are not operating. Therefore if a booster fan is in operation the streamers will be "up". This will permit the operators to diagnose a problem with the booster fan or with the booster fan supply damper.

The pressurization mode is the only automatically actuated mode change since bulk chlorine gas is no longer kept onsite and the chlorine monitors which previously initiated the recirculation mode have been de-activated.

The air entering the control room is continuously monitored by radiation detectors. One detector output above the setpoint will cause actuation of the pressurization mode.

A single train will pressurize the control room equal to or greater than 0.125 inches water gauge. The CRVS operation in maintaining the control room habitable is discussed in the FSAR, Section 9.4.1 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CRVS is designed in accordance with Seismic Category I requirements.

The CRVS is designed to maintain the control room environment for the duration of the most severe Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE
SAFETY ANALYSES

The CRVS components are arranged in redundant, safety related ventilation trains. The location of components and ducting ensures an adequate supply of filtered air to all areas requiring access. The CRVS provides airborne radiological protection for the control room operators, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in the FSAR, Chapter 15(Ref. 2).

The analysis of toxic gas releases demonstrates that the toxicity limits are not exceeded in the control room following a toxic chemical release, as presented in Reference 1.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The worst case single active failure of a component of the CRVS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CRVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant CRVS trains are required to be OPERABLE to ensure that at least one is available assuming a single failure disables the other train. The redundant train means a second train from the other unit (Ref. 5). Total system failure could result in exceeding a dose of 5 rem to the control room operator in the event of a large radioactive release.

The CRVS is considered OPERABLE when the individual components necessary to limit operator exposure are OPERABLE in both trains. A CRVS train is OPERABLE when the associated:

- a. main supply fan (one), filter booster fan (one) and pressurization fan (one) are OPERABLE;
- b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and
- c. Heaters, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.

APPLICABILITY

In MODES 1, 2, 3, 4, 5, and 6, and during movement of irradiated fuel assemblies CRVS must be OPERABLE to control operator exposure during and following a DBA or the release from the rupture of an outside waste gas tank.

During movement of irradiated fuel assemblies, the CRVS must be OPERABLE to cope with the release from a fuel handling accident.

CRVS OPERABILITY requires that for MODE 5 and 6 and during movement of irradiated fuel assemblies in either unit, when there is only one OPERABLE train of CRVS, the OPERABLE CRVS train must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which is energizing the OPERABLE CRVS train. This is an exception to LCO 3.0.6.

(continued)

BASES

ACTIONS

A.1

When one CRVS train is inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CRVS train is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CRVS train could result in loss of CRVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

CRVS OPERABILITY requires that for MODE 5 and 6 and during movement of irradiated fuel assemblies, when there is only one OPERABLE train of CRVS, the OPERABLE CRVS train must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which is energizing the OPERABLE CRVS train. This is an exception to LCO 3.0.6.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CRVS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, and C.2.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CRVS train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CRVS train in the recirculation mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected. As noted above, if only one CRVS train is OPERABLE, the OPERABLE train must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which is energizing the OPERABLE CRVS train. The power requirements for the one OPERABLE CRVS train assures that the ventilation function will not be lost during a fuel handling accident with a subsequent loss of off-site power.

An alternative to Required Action C.1 is to immediately suspend activities that could result in a release of radioactivity that

(continued)

BASES

ACTIONS

C.1, C.2.1, and C.2.2 (continued)

might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

D.1 and D.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CRVS trains inoperable, action must be taken immediately to suspend activities, including positive reactivity changes, that could result in a release of radioactivity that might enter the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CRVS trains are inoperable in MODE 1, 2, 3, or 4, the CRVS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE
REQUIREMENTS

Once actuated due to a fuel handling accident the CRVS must be protected against a single failure. This protection, although not required for immediate accident response, is assured by requiring that a backup power supply be provided as described above in Applicability. This back up is assured via the performance of non-TS surveillances that verify the ability to transfer power supplies.

The 31 day procedural verification of the separate vital power supplies for the redundant fans and the one hour operation of each supply, booster and pressurization supply fan (unless already operating) assures system reliability and two train redundancy.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized and operating automatically (filter temperature control). The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.10.2

This SR verifies that the required CRVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CRVS filter tests are in accordance with ANSI 510-1980 (Ref. 3). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

This SR verifies that each CRVS train automatically starts and operates in the pressurization mode on an actual or simulated actuation signal generated from a Phase "A" Isolation. The Frequency of 18 months is specified in ANSI 510-1980 (Ref. 3).

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CRVS. During the pressurization mode of operation, the CRVS is designed to pressurize the control room ≥ 0.125 inches water gauge positive pressure with respect to the outside atmosphere and adjacent areas in order to prevent unfiltered inleakage. The CRVS is designed to maintain this positive pressure with one train. The Frequency of 18 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).

(continued)

BASES

- REFERENCES
1. FSAR, Section 9.4.1.
 2. FSAR, Chapter 15.
 3. ANSI 510-1980.
 4. NUREG-0800, Section 6.4, Rev. 2, July 1981.
 5. DCM S-23F.
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B 3.7 PLANT SYSTEMS

B 3.7.11 Not Used

B 3.7 PLANT SYSTEMS

B 3.7.12 AUXILIARY BUILDING VENTILATION SYSTEM (ABVS)

BASES

BACKGROUND

The ABVS filters air from the area of the active ECCS components during the recirculation phase of a loss of coolant accident (LOCA). The ABVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area, if one of the pumps is operating, and the auxiliary building.

The ABVS consists of two trains. Each train is powered by a separate vital bus and consists of a supply fan and an exhaust fan. A single roughing and HEPA filter is common to both trains for normal operations and a single roughing filter, HEPA filter, and charcoal adsorber bank and a single manually initiated heater are common to both trains for emergency operations. Ductwork, valves or dampers, and instrumentation also form part of the system.

The ABVS has several modes of operation. These modes include: (1) Building Only; (2) Building and Safeguards; and (3) Safeguards only. In the Building Only mode of operation, the ABVS provides ventilation flow to all parts of the auxiliary building except for the ECCS pump rooms, but does take suction from the ECCS rooms. If any ECCS pump is started, the ABVS will automatically re-align to the Building and Safeguards mode of operation. In the Building and Safeguards mode of operation, ventilation is provided to the entire auxiliary building, including the ECCS pump rooms. In the Safeguards Only mode of operation, only the ECCS pump rooms and the lower reaches of the auxiliary building are provided with ventilation. If a SI signal is generated, the system will automatically realign such that all exhaust flow from the ECCS pump rooms passes through the common HEPA filter/charcoal adsorber bank prior to being exhausted to atmosphere. Whenever an SI signal is generated, the operator must manually energize the heater from the control room.

The ABVS is discussed in the FSAR, Sections 9.4 2, and 15.5 (Refs. 1, and 2, respectively) since it may be used for normal, as well as post accident, ventilation and atmospheric cleanup functions. The primary purpose of the single manually initiated heater is to maintain the relative humidity at an acceptable level, consistent with iodine removal efficiencies per ASTM D 3803-1989 (Ref. 3). There is no redundant heater since the failure of the charcoal absorber and heater train would constitute a second failure in addition to the RHR pump seal failure assumed in conjunction with a LOCA (Ref.7).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The design basis of the ABVS is established by the large break LOCA. The system evaluation assumes a passive failure of the ECCS outside containment, such as an RHR pump seal failure, during the recirculation mode. In such a case, the system limits radioactive release to within the 10 CFR 100 (Ref. 5) limits. The analysis of the effects and consequences of a large break LOCA is presented in Reference 2. The ABVS also functions, following a LOCA, in those cases where the ECCS goes into the recirculation mode of long term cooling, to clean up releases of smaller leaks, such as from valve stem packing.

The ventilation flow is also required to maintain the temperatures of the operating ECCS motors within allowable limits. The ventilation function has been designed for single failure and the system will continue to function to provide its ECCS motor cooling function.

Two types of system failures are considered in the accident analysis for radiation release: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal for any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.

The ABVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two trains of the ABVS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. Total system failure could result in the atmospheric release from the ECCS pump room exceeding 10 CFR 100 limits in the event of a Design Basis Accident (DBA).

ABVS is considered OPERABLE when the individual components necessary to maintain the ECCS pump room filtration and temperature are OPERABLE in both trains.

An ABVS train is considered OPERABLE when its associated:

- a. Supply and exhaust fans are OPERABLE;
- b. The common roughing filter, HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration functions; and
- c. A heater, ductwork, valves, and dampers are OPERABLE and air circulation can be maintained.

(continued)

BASES

APPLICABILITY In MODES 1, 2, 3, and 4, the ABVS is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODE 5 or 6, the ABVS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

ACTIONS

A.1

With the common HEPA filter and charcoal adsorber bank inoperable, the cooling function of the ABVS for ECCS motors is maintained; however, the filtration system function is lost. Since the entire function of the system is not lost, a 24 hour completion time is provided to restore the filters.

The 24 hour completion time is acceptable because it is a common filter system and the Completion Time is shorter than the ECCS Completion Time. The 24 hour Completion Time is based on the low probability of a DBA occurring during this time period.

B.1

With one ABVS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ABVS function.

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time). The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

Concurrent failure of two ABVS trains would result in the loss of both filtration and cooling capability; therefore, LCO 3.0.3 must be entered immediately.

C.1 and C.2

If the ABVS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. Monthly heater operations dry out any moisture that may have accumulated in the charcoal from humidity in the ambient air. Systems with heaters must be operated ≥ 10 continuous hours with the heaters energized and operating automatically (filter temperature control). Since the ABVS has only one common charcoal filter, one train needs to be operated for ≥ 10 hours to dry out the filter and the other train only needs to be operated long enough (≥ 15 minutes) to verify all components are operating correctly. Monthly verification of the separate OPERABLE vital power supplies for the exhaust fans, via a non-TS surveillance, assures system redundancy. The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required ABVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The ABVS filter tests are in accordance with References 3 and 4. The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR verifies that each ABVS train starts and operates on an actual or simulated actuation signal and that the system aligns to exhaust through the common HEPA filter and charcoal adsorber. The 18 month Frequency is consistent with that specified in References 3 and 4.

SR 3.7.12.4
Not Used.SR 3.7.12.5
Not Used.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.7.12.6

This SR verifies the leak tightness of dampers that isolate flow to the normally operating filter train. This SR assures that the flow from the auxiliary building passes through the HEPA filter and charcoal absorber unit when the ABVS Buildings and Safeguards or Safeguards Only modes have been actuated coincident with an SI. The 18 month Frequency is consistent with the requirements of Reference 4.

REFERENCES

1. FSAR, Section 9.4.2.
 2. FSAR, Section 15.5.
 3. ASTM D 3803-1989
 4. ANSI N510-1980
 5. 10 CFR 100.11.
 6. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
 7. DCM S-23B, "Main Auxiliary Building Heating and Ventilation System".
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B 3.7 PLANT SYSTEMS

B 3.7.13 Fuel Handling Building Ventilation System (FHBVS)

BASES

BACKGROUND

The FHBVS filters airborne radioactive particulates and radioactive iodine from the area of the fuel pool following a fuel handling accident. The FHBVS provides environmental control of temperature and humidity in the fuel pool area.

The FHBVS consists of two independent and redundant trains. Each train consists of, an exhaust prefilter, high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and an exhaust fan. A third non-vital exhaust fan is used for normal operation and has only a prefilter and a HEPA filter. Ductwork, valves or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal or loss of the normal exhaust fan E-4.

The FHBVS is a standby system, parts of which may also be operated during normal plant operations. Upon receipt of the actuating signal, normal air discharge from the fuel handling building is isolated and the normal exhaust fan shuts down and the vital exhaust fans start and the stream of ventilation air discharges through the system filter trains. The prefilter removes any large particles in the air, to prevent excessive loading of the HEPA filter and charcoal adsorber.

The FHBVS is discussed in the FSAR, Sections 9.4.4 and 15.5 (Refs. 1, and 2, respectively) because it may be used for normal, as well as post (fuel handling) accident, atmospheric cleanup functions.

APPLICABLE
SAFETY ANALYSES

The FHBVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that only one train of the FHBVS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident. In accordance with assumptions made in the fuel handling accident analysis, loss of offsite power is not considered concurrent with a fuel handling accident. However, loss of power is enveloped by the fuel handling accident

(continued)

BASES (continued)

APPLICABLE
SAFETY ANALYSES
(continued)

analysis. To maximize FHBVS capability to mitigate the consequences of a fuel handling accident, at least one of the FHBVS trains must be capable of being supplied from an operable emergency diesel generator at all times whenever fuel movement is taking place in the spent fuel pool. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 3).

The FHBVS satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant trains of the FHBVS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train, coincident with a loss of offsite power. This requires that when two trains of the FHBVS are OPERABLE, at least one train of the FHBVS must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which energizes the FHBVS train. When only one train is OPERABLE, an OPERABLE diesel generator must be directly associated with the bus which energizes that one OPERABLE FHBVS train. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR 100 (Ref. 4) limits in the event of a fuel handling accident.

The FHBVS is considered OPERABLE when the individual components necessary to control releases from fuel handling building are OPERABLE in both trains. An FHBVS train is considered OPERABLE when its associated:

- a. Exhaust fan is OPERABLE;
 - b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and
 - c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.
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APPLICABILITY

In MODE 1, 2, 3, 4, 5 or 6, the FHBVS is not required to be OPERABLE since it provides no safety function associated with these MODES of operation.

During movement of irradiated fuel in the fuel handling building, the FHBVS is required to be OPERABLE to alleviate the consequences of a fuel handling accident.

(continued)

BASES

ACTIONS

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

If moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

A.1, A.2 and A.3

With one FHBVS train inoperable, action must be taken to restore OPERABLE status immediately or place the remaining OPERABLE train in operation and verify that it has an OPERABLE emergency power source, or suspend movement of irradiated fuel assemblies in the fuel handling building. The suspension of movement of fuel assemblies does not preclude movement of assemblies to a safe position.

B.1 and B.2
Not Used.

C.1 and C.2
Not Used.

D.1

When two trains of the FHBVS are inoperable during movement of irradiated fuel assemblies in the fuel handling building, suspend movement of irradiated fuel assemblies in the fuel handling building. This does not preclude the movement of fuel assemblies to a safe position.

SURVEILLANCE
REQUIREMENTS

Once actuated due to a fuel handling accident the FHBVS must be protected against a single failure coincident with a loss of offsite power. Protection against a loss of power, although not required for immediate accident response, is assured by requiring that a backup power supply be provided as described above in the LCO section. This back up is assured via the performance of non-TS surveillances.

SR 3.7.13.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal

(continued)

BASES

SUREVILLANCE
REQUIREMENTS

SR 3.7.13.1 (continued)

operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

SR 3.7.13.2

This SR verifies that the required FHBVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The FHBVS filter tests are in accordance with (Ref. 5 and 6). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.13.3

This SR verifies that each FHBVS train starts and operates on an actual or simulated actuation signal and directs its exhaust flow through the HEPA Filters and charcoal absorber banks. The 18 month Frequency is consistent with Reference 6.

SR 3.7.13.4

This SR verifies the integrity of the fuel handling building enclosure. The ability of the fuel handling building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FHBVS. During the post accident mode of operation, the FHBVS is designed to maintain a slight negative pressure in the fuel handling building, to prevent unfiltered LEAKAGE. The FHBVS is designed to maintain the building pressure ≤ -0.125 inches water gauge with respect to atmospheric pressure. The Frequency of 18 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 7).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 6.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 9.4.4.
 2. FSAR, Section 15.5.
 3. Regulatory Guide 1.25.
 4. 10 CFR 100.
 5. ASTM D 3802-1989
 6. ANSI N510-1980.
 7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
 8. DCM S-23D, "Fuel handling Building HVAC System".
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B 3.7 PLANT SYSTEMS

B 3.7.14 Not Used

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the spent fuel pool design is given in the FSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section 15.4.5 and 15.5.22 (Ref. 3).

APPLICABLE
SAFETY ANALYSES

The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 100 (Ref. 5) limits.

According to Reference 4, there is 23 ft of water between the top of the damaged fuel rods and the fuel pool surface during a fuel handling accident. With 23 ft of water, the assumptions of Reference 4 can be used directly. Although there are other spent fuel pool elevations where fuel handling accidents can occur, the design basis fuel handling accident, which uses the conservative assumptions of RG 1.25, is expected to be bounding. To add conservatism, the analysis assumes that all fuel rods of the damaged fuel assembly fail.

In practice, the water level maintained for fuel handling provides more than 23 feet of water over the top of irradiated fuel assemblies seated in the storage racks. FSAR Section 9.1.4.3.4 requires the water level to be at or above 137 feet 8 inches during fuel handling to assure 8 feet of water shielding. This water level corresponds to 24 feet 6 inches above the top of the fuel assemblies in the racks and to 23 feet above a fuel assembly lying horizontally on top of the racks.

The spent fuel pool water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO The spent fuel pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists.

ACTIONS A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel pool is immediately suspended. This does not preclude movement of a fuel assemblies to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE
REQUIREMENTS SR 3.7.15.1

This SR is done during the movement of irradiated fuel assemblies as stated in the Applicability. This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.7.1.

(continued)

BASES

REFERENCES

1. FSAR, Section 9.1.2.
 2. FSAR, Section 9.1.3.
 3. FSAR, Section 9.1.4.3.4, 15.4.5 and 15.5.22.
 4. Regulatory Guide 1.25, Rev. 0.
 5. 10 CFR 100.11.
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B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

The spent fuel pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. Region 1, with 290 storage positions, has been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17.1. Region 2, with 1034 storage positions, has been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17.2.

The water in the spent fuel pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, require that the limiting k_{eff} is at or below the limit of 0.95 in the absence of soluble boron. Hence, the analysis of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time.

APPLICABLE
SAFETY ANALYSES

Most accident conditions result in negligible reactivity effect for either of the two regions (Ref. 2 and 3). However, scenarios can be postulated that could have more than a negligible positive reactivity effect. One such scenario is associated with placing a fuel assembly which is required to be stored in Region 1, in Region 2. This could potentially increase the k_{eff} of Region 2 above 0.95. Thus, to compensate for reductions in the subcriticality margin from postulated accident conditions, the spent fuel pool contains 2000 ppm soluble boron. The negative reactivity effect of the soluble boron more than compensates for the increased reactivity caused by the postulated accident scenarios. The accident analyses is provided in the FSAR, Section 15.5.22 (Ref. 4).

The concentration of dissolved boron in the spent fuel pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO The spent fuel pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in Reference 2, 3, and 4. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel pool.

APPLICABILITY This LCO applies whenever fuel assemblies are stored in the spent fuel pool.

ACTIONS A.1 and A.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies and immediately taking actions to restore the spent fuel pool boron concentration to greater than or equal to 2000 ppm. This suspension of fuel movement does not preclude movement of fuel assemblies to a safe position.

If the LCO is not met while moving fuel assemblies LCO 3.0.3 would not be applicable since the inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE
REQUIREMENTS SR 3.7.16.1

This SR verifies by chemical analysis that the concentration of boron in the spent fuel pool is at or above the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 31 day Frequency is appropriate because no major replenishment of pool water is expected to take place.

(continued)

BASES (continued)

REFERENCES

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).
 2. "Criticality Safety Evaluation of Region 1 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment," S.E.Turner, October 1993, Holtec Report HI-931076.
 3. "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment," S.E.Turner, October 1993, Holtec Report HI-931077.
 4. FSAR, Section 9.1 and 15.2.22.
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B 3.7 PLANT SYSTEMS

B 3.7.17 Spent Fuel Assembly Storage

BASES

BACKGROUND

The spent fuel storage pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. Region 1, with 290 storage positions, has been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17.1. Region 2, with 1034 storage positions, has been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17.2.

The water in the spent fuel pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, require that the limiting k_{eff} of the fuel configuration is at or below the limit of 0.95 in the absence of soluble boron. Hence, the analysis of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time.

Prior to movement of an assembly, it is necessary to verify that SR 3.7.16.1 is current.

APPLICABLE
SAFETY ANALYSES

The analyzed accidents that could have significant reactivity effects are the misplacement of a fuel assembly, a significant increase in spent fuel pool temperature above the design basis temperature of 150°F, or a cask drop accident (Ref. 2, 3, and 4). For these accident occurrences, the presence of soluble boron in the spent fuel storage pool (controlled by LCO 3.7.16, "Spent Fuel Pool Boron Concentration") ensures that k_{eff} will remain at or below 0.95.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

LCO The restrictions on the placement of fuel assemblies within the spent fuel pool, in accordance with LCO 3.7.17.1 and 3.7.17.2, ensures the k_{eff} of the spent fuel storage pool will always remain ≤ 0.95 , assuming the pool to be flooded with unborated water at a temperature of $\leq 150^{\circ}\text{F}$.

APPLICABILITY These LCOs apply whenever any fuel assembly is stored in the spent fuel pool.

ACTIONS A.1

The Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since the inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with LCO 3.7.17.1 and 3.7.17.2, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with LCO 3.7.17.1 and 3.7.17.2 which will return the fuel pool to an analyzed condition.

SURVEILLANCE
REQUIREMENTS SR 3.7.17.1.1 and SR 3.7.1.2

These SRs verify by administrative means that the each fuel assembly and its expected storage location are in accordance with the applicable LCO (3.7.17.1 or 3.7.17.2), prior to each fuel assembly move when the assembly is to be stored in Region 1 or 2 of the spent fuel pool.

REFERENCES 1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978 NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. FSAR, Section 15.5.22.3. "Criticality Safety Evaluation of Region 1 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment," S.E. Turner, October 1993, Holtec Report HI-931076.

3. Not Used.

4. "Criticality Safety Evaluation of Region 2 of the Diablo Canyon Spent Fuel Storage Racks with 5.0 % Enrichment," S.E. Turner, October 1993, Holtec Report HI-931076.

B 3.7 PLANT SYSTEMS

B 3.7.18 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 $\mu\text{Ci/gm}$ (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives, (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses. Operating at or below 0.1 $\mu\text{Ci/gm}$ ensures that in the event of a DBA, offsite doses will be less than 10 CFR 100 requirements.

APPLICABLE
SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the FSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed 10 CFR 100 limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generators are assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. The quantity of radioactivity released to the environment, due to an SGTR, depends upon primary and secondary coolant activity, iodine spiking effects, primary to secondary break flow flashing fractions, attenuation of iodine carried by the flashed portion of the break flow, partitioning of iodine between the liquid and steam phases, the mass of fluid released from the steam generator, and liquid-vapor partitioning in the condenser hotwell. All of these parameters were conservatively evaluated in a manner consistent with the recommendations of Standard Review Plan Section 15.6.3 and the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be $\leq 0.10 \mu\text{Ci/gm}$ DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.18.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.
 2. FSAR, Chapter 15.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC sources consist of offsite power sources (immediate access and delayed access), and the onsite standby power sources (three diesel generators (DGs) for each unit). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E AC Distribution System for each unit is divided into three load groups so that the loss of any one group does not prevent the minimum safety functions from being performed. Each load group has connections to two offsite power sources and a single DG.

Offsite power is supplied to the 230 kV and 500 kV switchyards from the transmission network by two 230 kV transmission lines and three 500 kV transmission lines. These two electrically and physically separated circuits provide AC power, through auxiliary and standby startup transformers, to the 4.16 kV ESF busses. A detailed description of the offsite power network and the circuits to the Class 1E buses is found in the FSAR, Chapter 8 (Ref. 2).

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite class 1E buses.

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System. Within 1 minute after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the load sequencer timers (auto transfer timers). Each individual timer connects a single ESF component.

The onsite standby power source for each 4.16 kV ESF bus is a dedicated DG. For Unit 1, DGs 1-1, 1-2, and 1-3 are dedicated to ESF buses H, G, and F, respectively. For Unit 2, DGs 2-1, 2-2, and 2-3 are dedicated to ESF buses G, H, and

(continued)

BASES

BACKGROUND
(continued)

F. A DG starts automatically on a safety injection (SI) signal (e.g., low pressurizer pressure or high containment pressure signals), undervoltage on the offsite standby startup source, or on an ESF bus degraded voltage or undervoltage signal (refer to LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation"). After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with an SI signal. The DGS will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone. Following the trip of offsite power, an undervoltage signal strips nonpermanent loads from the ESF bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the load sequencing timers (ESF timers). The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG. Each ESF component is provided with its own load sequencing timer.

In the event of a loss of preferred power, the ESF electrical loads are automatically connected to the DGs in sufficient time to provide for safe reactor shutdown and to mitigate the consequences of a Design Basis Accident (DBA) such as a loss of coolant accident (LOCA).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the DG in the process. Within 1 minute after the initiating signal is received, all loads needed to recover the unit or maintain it in a safe condition are returned to service.

Ratings for the six DGs satisfy the requirements of Regulatory Guide 1.9 (Ref.3). The continuous service rating of each DG is 2600 kW with 10% overload permissible for up to 2 hours in any 24 hour period. The ESF loads that are powered from the 4.16 kV ESF buses are listed in Reference 2.

Fuel oil is transferred from the storage tanks via the diesel fuel oil storage and transfer system to replenish the day tanks as required. The design incorporates sufficient redundancy so that a malfunction of either an active or a passive component will not impair the ability of the system to supply fuel oil. Two redundant fuel oil transfer pumps supply fuel oil to DG day tanks from either storage tank. One pump is adequate to supply the six DGs operating at full load. Each DG tank has two separate, redundant transfer pump start-stop level switches. Each level switch automatically starts a transfer pump and opens the supply header solenoid valve corresponding to the respective transfer pump, 0-1 or 0-2.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES

The initial conditions of DBA and transient analyses in the FSAR, Chapter 6 (Ref. 4) and Chapter 15 (Ref. 5), assume ESF systems are OPERABLE. The AC electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the AC electrical power sources is consistent with the initial assumptions of the Accident analyses and is based upon meeting the design basis of the unit. This results in maintaining at least one train of ESF systems powered by the onsite or offsite AC sources OPERABLE during Accident conditions in the event of:

- a. An assumed loss of all offsite power or all onsite AC power; and
- b. A worst case single failure.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two qualified circuits between the offsite transmission network and the onsite Class 1E Electrical Power System and separate and independent DGs for each Class 1E ESF bus ensure availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit.

Each offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the ESF buses.

The Unit 1 Offsite Circuit #1 consists of Startup Transformer 1-1 supplied from the immediate access 230 kV Switchyard power source, which feeds Startup Transformer 1-2 through series supply breakers 52VU12 and 52VU14. Startup Transformer 1-2 then supplies power through breaker 52HG15 to each vital bus feeder breaker (Bus F - 52HF14, Bus G - 52HG14, Bus H - 52HH14). The Unit 1 Offsite Circuit #2 is the delayed access 500 kV circuit which becomes available only after opening the motor operated disconnect to the main generator. This circuit consists of Auxiliary Transformer 1-2 supplied from the 500 kV Switchyard through the main bank

(continued)

BASES

LCO
(continued)

transformers. Auxiliary Transformer 1-2 supplies power directly to each of the vital bus feeder breakers (Bus F - 52HF13, Bus G - 52HG13, Bus H - 52HH13).

The Unit 2 Offsite Circuit #1 consists of Startup Transformer 2-1 supplied from the immediate access 230 kV Switchyard power source, which feeds Startup Transformer 2-2 through series supply breakers 52VU23 and 52VU24. Startup Transformer 2-2 then supplies power through breaker 52HG15 to each vital bus feeder breaker (Bus F - 52HF14, Bus G - 52HG14, Bus H - 52HH14). The Unit 2 Offsite Circuit #2 is a delayed access circuit which only becomes available after opening the motor operated disconnect to the main generator. This circuit consists of Auxiliary Transformer 2-2 supplied from the 500 kV Switchyard through the main bank transformers. Auxiliary Transformer 2-2 supplies power directly to each of the vital bus feeder breakers (Bus F - 52HF13, Bus G - 52HG13, Bus H - 52HH13).

Each DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby with the engine pre-lubed and pre-warmed. Additional DG capabilities must be demonstrated to meet required Surveillance, e.g., capability of the DG to automatically sequence the emergency loads onto the DG, following opening of the auxiliary breaker, on an ESF actuation signal while operating in parallel test mode.

The AC sources must be separate and independent (to the extent possible). For the DGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical. A circuit is normally connected to more than one ESF bus, with OPERABLE transfer capability to the other circuit, and does not violate separation criteria.

The two redundant diesel fuel oil supply trains supply fuel oil to DG day tanks from either storage tank. One supply train is adequate to supply the six DGs operating at full load.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

(continued)

BASES

APPLICABILITY
(continued)

- a. Acceptable fuel design, limits and reactor coolant pressure boundary limits are not exceeded as a result of AOs or abnormal transients; and
- b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources - Shutdown."

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered. If one required offsite circuit to only one Class 1E 4160 VAC bus is inoperable, SR 3.8.1.1 needs to be performed only for the affected Class 1E 4160 VAC bus. It is not necessary to perform SR 3.8.1.1 for the other Class 1E 4160 VAC buses.

The 230 kV system should be considered inoperable when the DCCP Shift Supervisor has been notified of system inoperability by the Diablo Canyon Switching Center, Grid Operations Scheduling, or Grid Shift Supervisor, in accordance with Transmission Operating Procedure 0-23, "Operating Instructions for Reliable Transmission Service for Diablo Canyon P.P."

A.2

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the on-site Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring

(continued)

BASES

ACTIONS

A.2 (continued)

during this period.

The second Completion Time for Required Action A.2 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 7 days. This could lead to a total of 10 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time; a DG could again become inoperable, the circuit restored OPERABLE, and an additional 7 days (for a total of 17 days) allowed prior to complete restoration of the LCO. The 10 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between 72 hour and 10 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition A was entered.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are powered from the three AC electrical power distribution subsystems (buses). Required features are redundant safety-related systems, subsystems, trains, components, and devices that

(continued)

BASES

ACTIONS

B.2 (continued)

depend on the diesel generators as a source of emergency power. Redundant required feature failures consist of inoperable features associated with one of the other Class 1E AC electrical power distribution subsystems, redundant to the subsystem associated with the inoperable DG. An example, if DG 1-1 (Bus H) were declared inoperable with safety injection pump 1-1 (Bus F) already inoperable. SIP 1-2 (Bus H) would then be required to be declared inoperable within 4 hours, and TS 3.0.3 entered. A Note has been added to point out that during operation in Modes 1, 2, and 3, two auxiliary feedwater pumps are required to meet the redundant features requirement to mitigate a feedwater line break. If both of the available AFW pumps are motor driven, neither of the two may be supplied by the DG which is inoperable. For example, declaring DG 1-1 inoperable would require maintaining the turbine driven AFW pump and motor driven AFW pump 1-2 OPERABLE, while declaring DG 1-2 inoperable would only require any 2 of the 3 AFW pumps OPERABLE.

The Completion Time for Required Action B.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

- a. An inoperable DG exists; and
- b. A required feature, redundant to a required feature associated with the inoperable DG on one of the other Class 1E AC electrical power distribution subsystems, is inoperable.

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable redundant required features associated with one of the OPERABLE DGs, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is Acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

In this Condition, the remaining two OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been

(continued)

BASES

ACTIONS

B.2 (continued)

lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

B.3.1 and B.3.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of OPERABLE DGs. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DGs, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on other DGs, the other DGs would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.3.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DGs, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of those DGs. If a DG has already started and loaded on a bus, it is not necessary to shutdown the DG and perform SR 3.8.1.2. The DG is verified OPERABLE since it is performing its intended function.

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.3.1 or B.3.2, the plant corrective action program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE DGs are not affected by the same problem as the inoperable DG.

B.4

Operation may continue in Condition B for a period that should not exceed 7 days. This AOT was revised from 72 hours to 7 days by License Amendment (LA) 44 for Unit 1 and LA 43 for Unit 2.

In Condition B, the remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 7 day Completion Time takes into account the capacity and capability of the

(continued)

BASES

ACTIONS

B.4 (continued)

remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable and that circuit is subsequently restored OPERABLE, the LCO may already have been not met for up to 72 hours. This could lead to a total of 10 days, since initial failure to meet the LCO, to restore the DG. At this time, an offsite circuit could again become inoperable, the DG restored OPERABLE, and an additional 72 hours (for a total of 13 days) allowed prior to complete restoration of the LCO. The 10 day Completion Time provides a limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 7 day and 10 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

As in Required Action B.2, the Completion Time allows for an exception to the normal "time zero" for beginning the allowed time "clock." This will result in establishing the "time zero" at the time that the LCO was initially not met, instead of at the time Condition B was entered.

C.1 and C.2

Required Action C.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The rationale for the reduction to 12 hours for Required Action C.1 is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not valid, and a shorter Completion Time of 12 hours is appropriate. Required features are redundant safety-related systems, subsystems, trains, components, and devices that depend on the DGs as a source of emergency power. These features are powered from the three Class 1E AC electrical power distribution subsystems. Examples of required features would include, but are not limited to, auxiliary saltwater pumps, centrifugal charging pumps, or motor-driven auxiliary

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

feedwater pumps.

The Completion Time for Required Action C.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

- a. All required offsite circuits are inoperable; and
- b. A required feature is inoperable.

If at any time during the existence of Condition C (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition C for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

- a. The configuration of the Class 1E AC electrical power system that remains available is not susceptible to a single bus or switching failure; and
- b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA transient. In fact, a simultaneous loss of offsite AC sources, a DBA, and a worst case single failure were postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a

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BASES

ACTIONS

C.1 and C.2 (continued)

period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

D.1 and D.2

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 12 hours.

In Condition D, individual redundancy is lost in the offsite electrical power system and may be lost in the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition C (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

E.1

With two or more DGs inoperable, the remaining onsite AC sources are inadequate. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system may be the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here

(continued)

BASES

ACTIONS

E.1 (continued)

is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Reference 6, with two or more DGS inoperable, operation may continue for a period that should not exceed 2 hours.

F.1

Condition F corresponds to a level of degradation in which one train of the DFO transfer system inoperable. The onsite AC electrical power systems are redundant and available to support ESF loads. However, one subsystem required for the onsite AC electrical system operability has lost its redundancy (DFO supply to the DGs).

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

G.1

With both trains of DFO inoperable, the onsite AC sources are inadequate (loss of DFO supply to all DGs). With an assumed loss of offsite electrical power, insufficient AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source for AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

H.1 and H.2

If the inoperable AC electric power sources cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the

(continued)

BASES

ACTIONS

H.1 and H.2 (continued)

required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

I.1

Condition I corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, further loss of the remaining offsite circuit will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

J.1

Condition J corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, further loss of a remaining DG will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

SURVEILLANCE
REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGS are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 9), and Regulatory Guide 1.137 (Ref. 10), as addressed in the FSAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of 3785 V is consistent with the second level undervoltage relay allowable values. This is the minimum steady state voltage needed on the 4160 volt vital buses to ensure adequate 4160 volt, 480 volt and 120 volt levels. The specified maximum steady state output voltage or 4400 V is equal to the maximum operating voltage for 4000 V motors specified in ANSI C84.1 (Ref.11). The maximum steady state output voltage of 4400 V ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating voltages. The specified minimum and maximum frequencies of

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BASESSURVEILLANCE
REQUIREMENTS
(continued)

the DG are 58.8 Hz and 61.2 Hz, respectively. These values are equal to $\pm 2\%$ of the 60 Hz nominal frequency and are derived from the recommendations given in Regulatory Guide 1.9 (Ref. 3).

SR 3.8.1.1

This SR ensures proper circuit continuity for the offsite AC electrical power supply to the onsite distribution network and availability of offsite AC electrical power. The breaker alignment verifies that each breaker is in its correct position to ensure that distribution buses and loads are connected to their preferred power source, and that appropriate independence of offsite circuits is maintained. The 7 day Frequency is adequate since breaker position is not likely to change without the operator being aware of it and because its status is displayed in the control room.

SR 3.8.1.2 and SR 3.8.1.7

These SRs help to ensure the availability of the standby electrical power supply to mitigate DBAs and transients and to maintain the unit in a safe shutdown condition.

To minimize the wear on moving parts that do not get lubricated when the engine is not running, these SRs are modified by a Note (Note 2 for SR 3.8.1.2) to indicate that all DG starts for these Surveillances may be preceded by an engine prelube period and followed by a warmup period prior to loading.

For the purposes of SR 3.8.1.2 and SR 3.8.1.7 testing, the DGS are started from standby conditions. Standby conditions for a DG means that the diesel engine coolant and oil are being continuously circulated and temperature is being maintained consistent with manufacturer recommendations. For the purposes of this SR, the diesel generator start will be initiated using one of the following signals: 1.) Manual, 2.) Simulated loss of offsite power, and 3.) Safety Injection actuation test signal.

In order to reduce stress and wear on diesel engines, some manufacturers recommend a modified start in which the starting speed of DGS is limited, warmup is limited to this lower speed, and the DGS are gradually accelerated to synchronous speed prior to loading. These start procedures are the intent of Note 3, which is only applicable when such modified start procedures are recommended by the manufacturer. Currently, the DGs are not able to gradually accelerate, and Note 3 does not apply. However, if the DG's governor is replaced with a governor with the ability to allow gradual acceleration, Note 3 may be applied.

SR 3.8.1.7 requires that, at a 184 day Frequency, the DG

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.1.2 and SR 3.8.1.7 (continued)

starts from standby conditions and achieves required speed within 10 seconds and required voltage and frequency within 13 seconds. The 10 second start requirement reflects the point during the DG's acceleration at which the DG is assumed to be able to accept load. The 13 second start requirement reflects the point at which the DG is assumed to have reached stable operation. The 10 and 13 second start requirements support the assumptions of the design basis LOCA analysis in the FSAR, Chapter 15 (Ref. 5).

The 10 and 13 second start requirements are not applicable to SR 3.8.1.2 (see Note 3) when a modified start procedure as described above is used. If a modified start is not used, the 10 and 13 second start requirements of SR 3.8.1.7 apply.

Since SR 3.8.1.7 requires a timed start, it is more restrictive than SR 3.8.1.2, and it may be performed in lieu of SR 3.8.1.2. This is the intent of Note 1 of SR 3.8.1.2.

The 31 day Frequency for SR 3.8.1.2 is consistent with Generic Letter 94-01 (Ref. 12). The 184 day Frequency for SR 3.8.1.7 is a reduction in cold testing consistent with Generic Letter 84-15 (Ref. 7). These Frequencies provide adequate assurance of DG OPERABILITY, while minimizing degradation resulting from testing.

SR 3.8.1.3

This Surveillance verifies that the DGs are capable of synchronizing with the offsite electrical system and accepting loads greater than or equal to the equivalent of the maximum expected accident loads. A minimum run time of 60 minutes is required to stabilize engine temperatures, while minimizing the time that the DG is connected to the offsite source.

Although no power factor requirements are established by this SR, the DG is normally operated at a power factor between 0.8 lagging and 1.0. The 0.8 value is the design rating of the machine, while the 1.0 is an operational limitation to ensure circulating currents are minimized. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The 31 day Frequency for this Surveillance is consistent with Regulatory Guide 1.9 (Ref. 3).

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that

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BASES

SURVEILLANCE
REQUIREMENTSSR 3.8.1.3 (continued)

mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time per unit in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank is a contained quantity sufficient for DG operation at full load for a nominal one-hour period. One hour is adequate time for an operator to take corrective action to restore the fuel oil supply to the affected day tank. The level is expressed as an equivalent volume in gallons.

The 31 day Frequency is adequate to assure that a sufficient supply of fuel oil is available, since the transfer pumps auto-starts are at a level above the minimum contained volume. Therefore, normal DG operation will not result in day tank level below the minimum required volume. Additional assurance of sufficient day tank contained volume is provided by a low level alarm.

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 10). This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from the fuel oil storage tanks to each day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed.

The Frequency of 31 days is adequate to verify proper operation of the fuel oil transfer pumps and day tank supply valves to maintain the required volume of fuel oil in the day tanks. The frequency has been proven acceptable through operating experience.

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Transfer of each 4.16 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit, which is the immediate access 230 kV, demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. Transfer of each 4.16 kV ESF bus power supply from the alternate offsite circuit (immediate access 230 kV) to the delayed access circuit (500 kV circuit) demonstrates the ability of the delayed access circuit. The 18 month Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This Note does not prohibit the application of LCO 3.0.5.

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single

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BASES

SURVEILLANCE
REQUIREMENTSSR 3.8.1.9 (continued)

load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. The single largest DG load is a centrifugal charging pump (CCP), which is rated at 600 hp. The CCP has a maximum demand, based on the maximum expected horsepower input and motor efficiency, of 515 kW. This Surveillance may be accomplished by:

- a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or
- b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The 2.4 seconds specified is equal to 60% of a typical 4 second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover following load rejection. The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9).

This SR is modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This Note does not prohibit the application of LCO 3.0.5.

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, Note 2 requires that, if synchronized to offsite power, testing must be performed using a power factor ≤ 0.9 lagging. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

SR 3.8.1.10

This Surveillance demonstrates the DG's capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.8.1.10 (continued)

tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG would experience following a full load rejection and verifies that the DG does not trip upon loss of the load. These acceptance criteria provide for DG damage protection. While the DG is not expected to experience this transient during an event and continue to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor \leq 0.87 lagging. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 9) and is intended to be consistent with expected fuel cycle lengths.

This SR has been modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbation to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This Note does not prohibit the application of LCO 3.0.5.

SR 3.8.1.11

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG autostart time of 10 seconds is derived from requirements of the accident analysis to respond to a design basis accident. The 10 second requirement reflects the assumption of the accident analysis that the DG has reached the point in its acceleration where the DG is able to accept load. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability is achieved. After

(continued)

BASES

SURVEILLANCE
REQUIREMENTSSR 3.8.1.11 (continued)

energization of the loads, steady state voltage and frequency are required to be within their limits.

The requirement to verify the connection and power supply of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. The permanently connected loads are the Class 1E 480 VAC buses. The permanently connected loads do not receive a load shed signal. In addition, the containment fan cooler units do not receive a load shed signal but are de-energized when their motor contactors drop out on undervoltage. The permanently connected loads are re-energized when the DG breaker closes to energize the bus. The auto-connected loads are those loads that are energized via their respective sequencing timer. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG systems to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGS during testing. For the purpose of this testing, the DGS must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This Note does not prohibit the application of LCO 3.0.5.

SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves stability by reaching the required voltage and frequency within the specified time (13 seconds) from the Safety Injection actuation signal and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.12.d and SR 3.8.1.12.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on a Safety Injection signal without loss of offsite power. The

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.1.12 (continued)

emergency loads are the ESF loads.

The requirement to verify the connection of permanent and autoconnected loads to the immediate access 230 kV offsite power system is intended to satisfactorily show the relationship of these loads to the DG loading logic. For a description of the permanent and auto-connected loads, see SR 3.8.1.11 Bases. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGS during testing. For the purpose of this testing, the DGS must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This Note does not prohibit the application of LCO 3.0.5.

SR 3.8.1.13

This Surveillance demonstrates that DG noncritical protective functions are bypassed when the diesel engine trip cutout switch is in the cutout position and the DG is aligned for automatic operation and critical protective functions (engine overspeed, generator differential current, and low lube oil pressure,) trip the DG to avert substantial damage to the DG unit. The noncritical trips include directional power, loss of field, breaker overcurrent, high jacket water temperature, and diesel overcrank. These noncritical trips are bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.1.13 (continued)

the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The 18 month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required DG from service. This Note does not prohibit the application of LCO 3.0:5.

SR 3.8.1.14

Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3), requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 2 hours of which is at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing must be performed using a power factor of ≤ 0.87 lagging. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate

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BASES

SURVEILLANCE
REQUIREMENTSSR 3.8.1.14 (continued)

the test. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This Note does not prohibit the application of LCO 3.0.5.

SR 3.8.1.15

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve stability by reaching the required voltage and frequency within 13 seconds. The 13 second time is derived from the requirements of the accident analysis to respond to a design basis accident. The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(5).

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least 2 hours at full load conditions prior to performance of this Surveillance is based on test data and manufacturer recommendations, which indicate 45 minutes is sufficient for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.16

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and load transfer from the DG to the offsite source can be made and the DG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open and can receive an auto close signal on bus undervoltage, and the load sequencing timers are reset.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.1.16 (continued)

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This Note does not prohibit the application of LCO 3.0.5.

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions will not be compromised as the result of testing. A Safety Injection signal, received while the DG is operating in a test mode, results in the auxiliary breaker opening and the emergency loads automatically sequencing onto the DG.

In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable.

This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The 18 month Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(8), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This Note does not prohibit the application of LCO 3.0.5.

SR 3.8.1.18

Under accident and loss of offsite power conditions, loads are sequentially connected to the bus by load sequencer timers. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The load sequence time interval tolerances ensure that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

With an ESF timer found to be outside the range of acceptable settings, the corresponding DG shall be declared inoperable in MODES 1, 2, 3, and 4, and the corresponding CONDITION

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.1.18 (continued)

followed. With an Auto Transfer timer found to be outside the range of acceptable settings, the corresponding DG shall be declared inoperable for all MODES.

The Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This Note does not prohibit the application of LCO 3.0.5.

SR 3.8.1.19

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with a Safety Injection signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGS during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for DGS. The reason for Note 2 is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This Note does not prohibit the application of LCO 3.0.5.

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REQUIREMENTS
(continued)

SR 3.8.1.20

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGS are started simultaneously.

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9).

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGS must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter 8.
3. Regulatory Guide 1.9, Rev. 3, July 1993.
4. FSAR, Chapter 6.
5. FSAR, Chapter 15.
6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
8. 10 CFR 50, Appendix A, GDC 18.
9. Regulatory Guide 1.108, Rev. 1, August 1977.
10. Regulatory Guide 1.137, Rev. 1, Oct 1979.
11. ASME, Boiler and Pressure Vessel Code, Section XI.
12. Generic Letter 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," May 31, 1994.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."

APPLICABLE SAFETY ANALYSES The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6. Worst case bounding events are deemed not credible in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted, provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

requirements are acceptable during shutdown modes based on:

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.
- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.
- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

In the event of an accident during shutdown, this LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with the Class 1E AC electrical power distribution subsystem required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es). Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the unit.

The Unit 1 Offsite Circuit #1 consists of Startup Transformer 1-1 supplied from the immediate access 230 kV Switchyard power source, which feeds Startup Transformer 1-2 through series supply breakers 52VU12 and 52VU14. Startup

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BASES

LCO
(continued)

Transformer 1-2 then supplies power through breaker 52HG15 to each vital bus feeder breaker (Bus F - 52HF14, Bus G - 52HG14, Bus H - 52HH14). The Unit 1 Offsite Circuit #2 is the delayed access 500 kV circuit which becomes available only after opening the motor operated disconnect to the main generator. This circuit consists of Auxiliary Transformer 1-2 supplied from the 500 kV Switchyard through the main bank transformers. Auxiliary Transformer 1-2 supplies power directly to each of the vital bus feeder breakers (Bus F - 52HF13, Bus G - 52HG13, Bus H - 52HH13).

The Unit 2 Offsite Circuit #1 consists of Startup Transformer 2-1 supplied from the immediate access 230 kV Switchyard power source, which feeds Startup Transformer 2-2 through series supply breakers 52VU23 and 52VU24. Startup Transformer 2-2 then supplies power through breaker 52HG15 to each vital bus feeder breaker (Bus F - 52HF14, Bus G - 52HG14, Bus H - 52HH14). The Unit 2 Offsite Circuit #2 is a delayed access circuit which only becomes available after opening the motor operated disconnect to the main generator. This circuit consists of Auxiliary Transformer 2-2 supplied from the 500 kV Switchyard through the main bank transformers. Auxiliary Transformer 2-2 supplies power directly to each of the vital bus feeder breakers (Bus F - 52HF13, Bus G - 52HG13, Bus H - 52HH13).

The DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses.

With administrative controls in place, it is acceptable for Class 1E AC electrical power distribution subsystems to be cross tied during shutdown conditions, allowing a single offsite power circuit or a single DG to supply the required Class 1E AC electrical power distribution subsystems.

The two redundant diesel fuel oil transfer pumps supply fuel oil to DG day tanks from either storage tank. One pump is adequate to supply the six DGs operating at full load. Only one train is required to be OPERABLE in MODEs 5 or 6.

APPLICABILITY

The AC sources required to be OPERABLE in MODEs 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

- a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;

(continued)

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APPLICABILITY
(continued)

- b. Systems needed to mitigate a fuel handling accident are available;
- c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1

An offsite circuit would be considered inoperable if it were not available to the required Class 1E bus(es). If two Class 1E AC electrical power distribution subsystems are required by LCO 3.8.10, and one Class 1E AC electrical power distribution subsystem has offsite power available, the remaining Class 1E AC electrical power distribution subsystem may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required AC electrical power distribution subsystems, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions. The Required Action to

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4 (continued)

suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory provided the required SDM is maintained.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not an AC electrical power distribution subsystem is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a de-energized AC electrical power distribution subsystem.

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SR 3.8.2.1

SR 3.8.2.1 lists the SRs from LCO 3.8.1 that are applicable for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.8 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.17 is not required to be met because the required OPERABLE DG is not required to undergo periods of being synchronized to the offsite circuit. SR 3.8.1.20 is excepted because starting independence is not required with the DG that is not required to be operable. SR 3.8.1.12, SR 3.8.1.13, SR 3.8.1.18 (for ESF timers), and SR 3.8.1.19 are excepted because SI response functions are not required to be operable.

This SR is modified by a Note listing the applicable SRs from LCO 3.8.1 that are not required to be performed. The reason

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BASES

SURVEILLANCE SR 3.8.2.1 (continued)

for the Note is to preclude requiring the OPERABLE DG from being paralleled with the offsite power network or otherwise rendered inoperable during performance of an SR. The note would also preclude deenergizing a required 4160 V ESF bus or disconnecting a required offsite circuit for performance of an SR. With limited AC sources available, a single event could compromise both the required circuit and the DG. The note does not except the requirement for the DG, 4160 V ESF bus, or offsite circuit to be capable of performing the particular function, just that the capability of need not be demonstrated while that source of power is being relied on to support meeting the LCO.

Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, Starting Air, and Turbocharger Air Assist

BASES

BACKGROUND

The diesel fuel oil storage system consists of two common tanks with a nominal capacity of 40,000 gallons each. The TS-required fuel oil quantity is based on the calculated fuel oil consumption necessary to support the operation of the DGs to power the minimum engineered safety feature (ESF) systems required to mitigate a design basis accident (LOCA) in one unit and those minimum required systems for a concurrent non-LOCA safe shutdown in the remaining unit (both units initially in Mode 1 operation). The fuel oil consumption is calculated for a period of 7 days operation of minimum ESF systems. This requirement provides a sufficient operating period within which offsite power can be restored and/or additional fuel can be delivered to the site.

Fuel oil is transferred from the storage tanks via the diesel fuel oil storage and transfer system to replenish the day tanks as required. The design incorporates sufficient redundancy so that a malfunction of either an active or a passive component will not impair the ability of the system to supply fuel oil. Two redundant fuel oil transfer pumps supply fuel oil to DG day tanks from either storage tank. One pump is adequate to supply the six DGs operating at full load. Each DG tank has two separate, redundant transfer pump start-stop level switches. Each level switch automatically starts a transfer pump and opens the supply header solenoid valve corresponding to the respective transfer pump, 0-1 or 0-2. In addition, high and low level alarms are provided on each day tank and activate alarms both locally and in the control room.

The diesel lube oil storage requirement is based upon a conservative usage factor of 1% of fuel oil consumption. The storage system used to meet this requirement is that located in the warehouse where 650 gallons of lube oil is stored in drums. This storage is augmented by a second storage location within the diesel engine itself. The lube oil level on each engine's dip stick is maintained 5 inches above the engine's operability limit. This provides approximately 120 gallons of usable lube oil within each of the 6 diesel engines.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS

The DG fuel oil consumption is calculated for a period of 7 days operation of minimum ESF systems. This requirement provides a sufficient operating period within which offsite power can be restored and/or additional fuel can be delivered to the site.

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. The total engine oil sump inventory (all engines) is capable of supporting a minimum of 7 days of operation at minimum ESF loads. The onsite storage inventory (warehouse) is in addition to the engine oil sump also sufficient to ensure 7 days of continuous operation. These supplies are sufficient to allow the operators to replenish lube oil from outside sources as a third resource.

Each DG has two redundant 100% capacity air start systems and a turbocharger air assist system with adequate capacity for three successive start attempts each on the DG without recharging the air start receivers or the turbocharger air assist air receiver.

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 4), and in the FSAR, Chapter 15 (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, air start, and turbocharger air assist subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of minimum ESF systems operation. The required combined stored diesel fuel oil is a contained quantity with different storage requirements for unit operation in MODE 1, 2, 3, and 4 and for MODE 5 and 6. With both units operating in MODE 1, 2, 3, and 4, the required level is $\geq 65,000$ gallons. With one unit operating in MODE 1, 2, 3, or 4, and the other unit in MODE 5 or 6, the required fuel oil level is 33,000 gallons plus 26,000 gallons, for a total of 59,000 gallons combined storage. With both units in MODE 5 or 6.

(continued)

BASES

LCO
(continued)

the required fuel oil level is 52,000 gallons. The required combined stored fuel oil was revised by License Amendment 74 for Unit 1 and 73 for Unit 2.

The Note permits diesel fuel oil storage tank cleaning to be performed. Each tank is required to be cleaned on a 10-year frequency. Conducting the cleaning requires the tank to be taken out of service. For this infrequent event, the inventory in the remaining tank is sufficient to support operation of the DGs to power the minimum required loads to maintain safe conditions for a period of 4 days, considering one unit in MODE 1, 2, 3, 4, 5, or 6 and one unit in MODE 6 with 23 feet of water above the reactor vessel flange or with the reactor vessel defueled. The requirements for diesel fuel oil tank cleaning were approved by License Amendment 74 for Unit 1 and 73 for Unit 2.

The fuel oil is also required to meet specific standards for quality. Additionally, sufficient lubricating oil supply must be available to ensure the capability to operate at full load for 7 days. This requirement, in conjunction with an ability to obtain replacement supplies within 7 days, supports the availability of DGs required to shut down the reactor and to maintain it in a safe condition for an anticipated operational occurrence (A00) or a postulated DBA with loss of offsite power. DG day tank fuel requirements, as well as transfer capability from the storage tank to the day tank, are addressed in LCO 3.8.1, "AC Sources - Operating," and LCO 3.8.2, "AC Sources - Shutdown."

The starting air system and turbocharger air assist system are required to have a minimum capacity for three successive DG start attempts without recharging the air start receivers or the turbocharger air assist air receiver.

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an A00 or a postulated DBA. Since stored diesel fuel oil, lube oil, starting air, and turbocharger air assist subsystems support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil, starting air, and turbocharger air assist are required to be within limits when the associated DGs are required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG or diesel fuel oil storage tank, except for Condition A. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one

(continued)

BASES

ACTIONS
(continued)

inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) are governed by separate Condition entry and application of associated Required Actions.

Condition A is excepted from this allowance for diesel fuel oil storage tanks, since the requirement is for a combined storage quantity contained in both storage tanks. However, the Note would still allow separate Condition entry into a DG subsystem's Required Action coincident with Condition A.

A.1 and A.2

In this Condition, the 7 day fuel oil supply for the DGs is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6 day supply. These circumstances may be caused by events, such as full load operation required after an inadvertent start while at minimum required level, or feed and bleed operations, which may be necessitated by increasing particulate levels or any number of other oil quality degradations. This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the associated DGs inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period. Should the specified 6 day fuel oil supply for both units not be available, but the available supply is still greater than that required to support operation of one unit, then that available supply can be allocated to a selected unit, and the DGs declared inoperable under Action H need only be the ones associated with the unit that has the inadequate supply.

B.1

With diesel engine lube oil stored inventory < 650 gal, sufficient lubricating oil to support 7 days of continuous DG operation based on minimum 7 day ESF systems loading at 1% of fuel oil consumption. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6 day supply of 610 gallons. This ACTION should be entered based upon warehouse inventory of less than 650 gallons with both units in MODES 1, 2, 3 or 4 and less than 590 gallos with one unit in MODES 1, 2, 3, or 4 and the other in MODES 5 or 6. This restriction allows sufficient time to obtain the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures

(continued)

BASES

ACTIONS

B.1 (continued)

will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion of SR 3.8.3.3. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, and particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DGs inoperable. The 7 day Completion Time allows for further evaluation, re-sampling and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3. 3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or combinations of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is a high likelihood that the DG would still be capable of performing its intended function.

E.1

With both starting air receiver pressures < 180 psig, sufficient capacity for three successive DG start attempts does not exist. However, as long as one receiver pressure is > 150 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while one air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining air start capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

(continued)

BASES

ACTIONS
(continued)

F.1

With turbocharger air assist air receiver pressure < 180 psig, sufficient capacity for three successive DG start attempts does not exist. However, as long as the receiver pressure is > 150 psig, there is adequate capacity for at least one start attempt, and the DG can be considered OPERABLE while the turbo air assist air receiver pressure is restored to the required limit. A period of 48 hours is considered sufficient to complete restoration to the required pressure prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity, the fact that most DG starts are accomplished on the first attempt, and the low probability of an event during this brief period.

G.1

With a Required Action and associated Completion Time not met, or one or more DG's, lube oil, starting air, or turbocharger air assist subsystem not within limits for reasons other than addressed by Conditions B, E, or F, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

H.1, H.2, and H.3

With a Required Action and associated Completion Time not met, or the fuel oil storage tanks not within limits for reasons other than addressed by Conditions A, C, or D, the fuel oil storage tanks may be incapable of supporting the DGs in performing their intended function. This condition requires declaring inoperable, all the DGs on the unit(s) associated with either the inadequate fuel oil inventory, the fuel storage tank(s) having particulate outside the limit, and/or the fuel storage tank(s) having properties outside limits; and shutting down to MODE 3 in 6 hours and MODE 5 in 36 hours any associated unit(s) operating in MODE 1,2,3, or 4.

SURVEILLANCE
REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support DG operation for 7 days based a realistic (minimum) ESF systems loading profile. The 7 day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.3.2 (continued)

This Surveillance ensures that sufficient lube oil inventory is available to support at least 7 days of operation for each DG at minimum ESF systems loading. The 650 gal requirement is based on the DG manufacturer consumption values for the run time of the DG at 1% of fuel oil consumption. The storage system used to meet this requirement is that located within the warehouse where 650 gallon of lube oil is stored in drums.

A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the unit staff.

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the storage tanks. The tests, limits, and applicable ASTM Standards are as follows:

- a. Sample the new fuel oil in accordance with ASTM D4057-81 (Ref. 6);
- b. Verify in accordance with the tests specified in ASTM D975-81 (Ref. 6) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60°F of $\geq 27^\circ$ and $\leq 39^\circ$, a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point of $\geq 125^\circ\text{F}$; and
- c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176- or a water and sediment content of ≤ 0.05 volume percent when tested in accordance with ASTM D-1796-83 (Ref. 6).

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO concern since the fuel oil is not added to the storage tanks.

Within 30 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975- 81 (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975- 81 (Ref. 6), except that the analysis for sulfur

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.3.3 (continued)

may be performed in accordance with ASTM D1552-79 (Ref. 6) or ASTM D2622-82 (Ref. 6). The 30 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

If the analysis of the new fuel oil sample indicates that one or more of the other properties specified in Table 1 of ASTM D975-81 are not within limits, then Required Action D.1 shall be entered, allowing 30 days to restore fuel oil properties to within limits.

Fuel oil degradation during long term storage shows up as an increase in particulates, due mostly to oxidation. The presence of particulates does not mean the fuel oil will not burn properly in a diesel engine. The particulates can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D2276-78, Method A (Ref. 6). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Each tank must be considered and tested separately. ASTM D 2276-78 was written specifically for aviation fuel. However, it is used in this SR to evaluate diesel fuel oil. Therefore, it may be necessary to perform this test as a modified method. For example, a 500 ml sample may be analyzed rather than a one gallon sample.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of six engine start cycles without recharging. Each start cycle is 15 seconds of cranking. The pressure specified in this SR is intended to reflect the lowest value at which three starts can be accomplished.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.3.4 (continued)

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, or from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

SR 3.8.3.6

This Surveillance ensures that, without the aid of the refill compressor, sufficient turbocharger air assist air receiver capacity for each DG is available. The system design requirements provide for a minimum of six engine start cycles without recharging. Each start cycle is 15 seconds of cranking. The pressure specified in this SR is intended to reflect the lowest value at which three starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal turbocharger air assist air receiver pressure.

REFERENCES

1. FSAR, Section 9.5.4.2.
2. Regulatory Guide 1.137.
3. ANSI N195-1976, Appendix B.

(continued)

BASES

4. FSAR, Chapter 6.
 5. FSAR, Chapter 15.
 6. ASTM Standards: D4057-[81; D975- 81; D4176- 82;
D1796-83; D1552- 79; D2622- 82; D2276-78, Method A.
 7. ASTM Standards, D975, Table 1.
 8. ASME, Boiler and Presser Vessel Code, Section XI.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 DC Sources - Operating

BASES

BACKGROUND

The Class 1E DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and backup 120 VAC vital bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the Class 1E DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 125 VDC electrical power system consists of three independent safety related Class 1E DC electrical power subsystems. Each subsystem consists of one 60-cell 125 VDC battery (Batteries 11(21), 12 (22), and 13 (23)), the dedicated battery charger and backup charger for each battery, and all the associated switchgear, control equipment, and interconnecting cabling.

There are two backup chargers for the three Class 1E DC subsystems. One backup charger is shared between two Class 1E DC subsystems. The other backup charger is dedicated to the third Class 1E DC subsystem. The backup chargers provide backup service in the event that the preferred battery charger is out of service. If the backup battery charger is substituted for one of the preferred battery chargers, then the requirements of independence and redundancy between subsystems are not maintained, and operation in this condition is limited to 14 days by Condition B.

During normal operation, the 125 VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

The DC electrical power subsystems provide the control power for its associated Class 1E AC power load group, 4.16 kV switchgear, and 480V load centers. The DC electrical power subsystems also provide DC electrical power to the inverters, which in turn are backup sources to power the 120 VAC vital buses.

The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution System - Operating," and LCO 3.8.10, "Distribution Systems - Shutdown."

(continued)

BASES

BACKGROUND
(continued)

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours as discussed in the FSAR, Chapter 8 (Ref. 4).

Each 125 VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem.

The batteries for the three DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery. The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

Each DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 12 hours while supplying normal steady state loads discussed in the FSAR, Chapter 8 (Ref. 4).

APPLICABLE
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 6), and in the FSAR, Chapter 15 (Ref. 7), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite AC power or all onsite AC power; and
- b. A worst case single failure.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES

LCO

The DC electrical power subsystems, each subsystem consisting of one battery, battery charger for each battery and the corresponding control equipment and interconnecting cabling supplying power to the associated bus are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (A00) or a postulated DBA. Loss of any one DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE DC electrical power subsystem requires the battery and its normal or backup charger to be operating and connected to the associated DC bus.

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of A00s or abnormal transients; and
- b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

The DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.5, "DC Sources - Shutdown."

ACTIONS

A.1

Condition A represents one Class 1E DC electrical power subsystem and associated ESF equipment with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected subsystem. The 2 hour limit is consistent with the allowed time for an inoperable DC distribution subsystem.

If one of the required DC electrical power subsystems is inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst case single failure would, however, result in the loss of

(continued)

BASES
ACTIONSA.1 (continued)

one of the two remaining 125 VDC electrical power subsystems with attendant loss of ESF functions, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 8) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

B.1

The design of the 125 VDC electrical power distribution system is such that a battery can have associated with it a full capacity charger powered from its associated 480 VAC vital bus or an alternate full capacity charger powered from another 480 VAC vital bus. However, operation in the latter condition or, with two chargers powered by the same vital bus is limited to 14 days.

C.1 and C.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 8).

SURVEILLANCE
REQUIREMENTSSR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations. The 31 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 9).

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.8.4.2

Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each intercell, interrack, and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The resistance of cell-to-cell connecting cables does not have to be included in measurement of connection resistance.

The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.

SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The presence of physical deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The 18 month Frequency for this SR is based on operational experience related to battery integrity and physical degradation.

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of intercell, interrack, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection. The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR provided visible corrosion is removed during performance of SR 3.8.4.4. The resistance of cell-to-cell connecting cables does not have to be included in measurement of connection resistance for SR 3.8.4.5.

The Surveillance Frequencies of 18 months is based on operational experience related to corrosion and connection resistance trends.

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BASES

SURVEILLANCE
REQUIREMENTS
(continued)SR 3.8.4.6

This SR requires that each battery charger be capable of supplying 400 amps at ≥ 130 V for ≥ 4 hours. These requirements are based on the design capacity of the chargers (Ref. 4). According to Regulatory Guide 1.32 (Ref. 10), the battery charger supply is required to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensures that these requirements can be satisfied.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 18 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.8.4.7

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements.

The Surveillance Frequency of 18 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 10) and Regulatory Guide 1.129 (Ref. 11), which state that the battery service test should be performed during refueling operations or at some other outage, with intervals between tests, not to exceed 18 months.

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the one minute duration of time equal to that of the service test.

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BASES

SURVEILLANCE
REQUIREMENTSSR 3.8.4.7 (continued)

A modified discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test. The modified performance discharge test and service test should be performed in accordance with IEEE-450 (Ref. 9).

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This Note does not prohibit the application of LCO 3.0.5.

SR 3.8.4.8

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

A battery modified performance discharge test is described in the Bases for SR 3.8.4.7. Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.4.8; however, only the modified performance discharge test may be used to satisfy SR 3.8.4.8 while satisfying the requirements of SR 3.8.4.7 at the same time.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 9) and IEEE-485 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected service life, the Surveillance Frequency is reduced to 18 months. Degradation is indicated, according to IEEE-450 (Ref. 9), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is < 90% of the manufacturer's rating. The Surveillance Frequency basis is consistent with IEEE-450 (Ref. 9), except if accelerated testing is required, it will be performed at an 18-month frequency to coincide with a refueling outage.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.4.8 (continued)

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This Note does not prohibit the application of LCO 3.0.5.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
 2. Regulatory Guide 1.6, March 10, 1971.
 3. IEEE-308-1978.
 4. FSAR, Chapter 8.
 5. IEEE-485- 1978, June 1983.
 6. FSAR, Chapter 6.
 7. FSAR, Chapter 15.
 8. Regulatory Guide 1.93, December 1974.
 9. IEEE-450- 1995.
 10. Regulatory Guide 1.32, February 1977.
 11. Regulatory Guide 1.129, December 1974.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."

APPLICABLE SAFETY ANALYSES The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The DC sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO The DC electrical power subsystems, each subsystem consisting of one battery, one battery charger per battery, and the corresponding control equipment and interconnecting class 1E cabling within the subsystem, are required to be OPERABLE to support required trains of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown." An OPERABLE subsystem consists of a DC bus connected to a battery with an OPERABLE battery charger which is fed from an OPERABLE AC vital bus. The OPERABLE AC vital bus must have an OPERABLE DG capable of starting and automatic loading in the event of a LOOP. This ensures that the DC bus and battery

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BASES

LCO
(continued)

will be powered by a battery charger in the event of a LOOP. With administrative controls in place, DC buses may be cross-tied when a battery is taken out for maintenance provided that the battery and the Class 1E cross-tie has sufficient capacity and protection for its own loads and the cross-tie loads. The resulting circuit is not required to be single failure resistant. This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

- a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core;
- b. Required features needed to mitigate a fuel handling accident are available;
- c. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

One or more required DC electrical power subsystems may be inoperable provided that the remaining OPERABLE DC electrical

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BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

power subsystem(s) support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown," and are capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.8. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of an SR. This note does not except the requirement for the battery to be capable of performing the particular function, just that the capability need not be demonstrated while that source of power is being relied on to meet the LCO.

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BASES

REFERENCES

1. FSAR, Chapter 6.
 2. FSAR, Chapter 15.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Battery Cell Parameters

BASES

BACKGROUND This LCO delineates the limits on electrolyte temperature, level, float voltage, and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.4, "DC Sources - Operating," and LCO 3.8.5, "DC Sources - Shutdown."

APPLICABLE SAFETY ANALYSES The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the required DC electrical power subsystem(s) OPERABLE during accident conditions, in the event of:

- a. An assumed loss of all offsite AC power or all onsite AC power; and
- b. A worst case single failure.

Battery cell parameters satisfy the Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO Battery cell parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Electrolyte limits are conservatively established, allowing continued DC electrical system function even with Category A and B limits not met.

APPLICABILITY The battery cell parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery OPERABILITY is only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.4 and LCO 3.8.5.

(continued)

BASES

ACTIONS

A.1, A.2, and A.3

With one or more cells in one or more batteries not within limits (i.e., Category A limits not met, Category B limits not met, or Category A and B limits not met) but within the Category C limits specified in Table 3.8.6-1 in the accompanying LCO, the battery is degraded but there is still sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of Category A or B limits not met and operation is permitted for a limited period.

The pilot cell electrolyte level and float voltage are required to be verified to meet the Category C limits within 1 hour (Required Action A.1). This check will provide a quick indication of the status of the remainder of the battery cells. One hour provides time to verify the electrolyte level and to confirm the float voltage of the pilot cells. One hour is considered a reasonable amount of time to perform the required verification.

Verification that the Category C limits are met (Required Action A.2) provides assurance that during the time needed to restore the parameters to the Category A and B limits, the battery is still capable of performing its intended function. A period of 24 hours is allowed to complete the initial verification because specific gravity measurements must be obtained for each connected cell. Taking into consideration both the time required to perform the required verification and the assurance that the battery cell parameters are not severely degraded, this time is considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A or B limits. This periodic verification is consistent with the normal Frequency of pilot cell Surveillances.

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the Required Actions of Condition A within

(continued)

BASES

ACTIONS

B.1 (continued)

the required Completion Time or average electrolyte temperature of representative cells less than 60°F, are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

SURVEILLANCE
REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters on a 31-day frequency are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of pilot cells.

SR 3.8.6.2

The quarterly inspection of specific gravity is more conservative than IEEE-450 (Ref. 3), which requires a yearly frequency. In addition, within 7 days of a battery discharge < 118 V or a battery overcharge > 145 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to 118 V, do not constitute a battery discharge provided the battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells is $\geq 60^\circ\text{F}$, is consistent with a recommendation of IEEE-450 (Ref. 3), that states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on battery sizing calculations.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for the

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BASES

SURVEILLANCE
REQUIREMENTS

Table 3.8.6-1 (continued)

designated pilot cell in each battery. The cell selected as the pilot cell is that whose temperature, voltage, and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 3), with the extra $\frac{1}{4}$ inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote a to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage, and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 (Ref. 3) recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the recommendations of IEEE-450 (Ref. 3), which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.195 (0.015 below the manufacturer minimum fully charged specific gravity or a battery charging current that had stabilized at a low value). This value is characteristic of a charged cell with adequate capacity. According to IEEE-450 (Ref. 3), the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature and level. The correction factors are provided by the battery manufacturer. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the minimum normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.190 (0.020 below the manufacturer minimum fully charged specific gravity) with the average of all connected cells > 1.200 (0.010 below the manufacturer minimum fully charged specific gravity). These values are based on manufacturer's recommendations. The minimum specific gravity value

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BASES

SURVEILLANCE
REQUIREMENTS

Table 3.8.6-1 (continued)

required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

Category C defines the minimum allowable limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists, and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limits for float voltage is based on IEEE-450 (Ref. 3), which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit of average specific gravity ≥ 1.190 is based on manufacturer recommendations (0.020 below the manufacturer recommended minimum fully charged specific gravity). In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

The footnotes to Table 3.8.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 3.8.6-1 requires the above mentioned correction for electrolyte level and temperature, with the exception that level correction is not required when battery charging current is < 2 amps on float charge. This current provides, in general, an indication of a battery in a charged condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 3). Footnote (c) to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days, each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep

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BASES

SURVEILLANCE
REQUIREMENTS

Table 3.8.6-1 (continued)

discharge) specific gravity gradients are not significant,
and confirming measurements may be made in less than 7 days.

REFERENCES

1. FSAR, Chapter 6.
 2. FSAR, Chapter 15.
 3. IEEE-450- 1995.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Inverters - Operating

BASES

BACKGROUND

The Class 1E UPS inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The function of the inverter is to provide AC electrical power to the vital buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station battery provides an uninterruptible power source for the instrumentation and controls for the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS). Specific details on inverters and their operating characteristics are found in the FSAR, Chapter 7 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume Engineered Safety Feature systems are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC vital buses OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; and
- b. A worst case single failure.

Inverters are a part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The Class 1E UPS inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

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BASES

LCO
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Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained. The four inverters ensure an uninterruptible supply of AC electrical power to the 120 VAC vital buses even if the 4.16 kV safety buses are de-energized.

Operable inverters require the associated 120 VAC vital bus to be powered by the inverter with output voltage within tolerances, and power input to the inverter from a 125 VDC station battery. Alternatively, power supply may be from an internal AC source via rectifier as long as the station battery is available as the uninterruptible power supply.

APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.8, "Inverters - Shutdown."

ACTIONS

A.1

With a required inverter inoperable, its associated 120 VAC vital bus becomes inoperable until it is re-energized from its Class 1E constant voltage source transformer.

For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures that the 120 VAC bus is re-energized within 2 hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the 120 VAC vital bus is powered from its constant voltage source, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to

(continued)

BASES

ACTIONS

A.1 (continued)

the 120 VAC vital buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and 120 VAC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. FSAR, Chapter 7.
 2. FSAR, Chapter 6.
 3. FSAR, Chapter 15.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The Class 1E UPS inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum inverters to each 120 VAC vital bus during MODES 5 and 6 ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

The inverters were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO The Class 1E UPS inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The battery powered inverters provide uninterruptible supply of AC electrical power to the 120 VAC vital buses even if the 4.16 kV safety buses are de-energized. OPERABILITY of the Class 1E 120 VAC vital bus requires that the 120 VAC

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BASES

LCO
(continued)

vital bus be powered by the inverter. An OPERABLE Class 1E UPS inverter is one that is connected to an OPERABLE DC subsystem (see B 3.8.5). The resulting circuit is not required to be single failure resistant. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The inverters required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

- a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;
- b. Systems needed to mitigate a fuel handling accident are available;
- c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

One or more Class 1E UPS inverters may be inoperable provided that the remaining OPERABLE inverters support the Class 1E 120 VAC vital bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown," and are capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS,

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BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare required features inoperable with the associated Class 1E UPS inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required Class 1E UPS inverters and to continue this action until restoration is accomplished in order to provide the necessary Class 1E UPS inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer.

SURVEILLANCE
REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage output ensures that the required power is readily available for the instrumentation connected to the 120 VAC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

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BASES

REFERENCES

1. FSAR, Chapter 6.
 2. FSAR, Chapter 15.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.9 Distribution Systems - Operating

BASES

BACKGROUND

The onsite Class 1E electrical power distribution system is designed with three 4160 V and 480 V vital buses (F, G, and H) and three 125 V DC vital buses. The plant protection system (PPS) is designed with four input channels (I, II, III, and IV) powered from four 120 VAC vital buses (1, 2, 3, and 4). The four channels provide input to the solid state protection system (SSPS) Trains A and B. Each SSPS train actuates engineered safety feature (ESF) equipment in the three vital AC and DC buses and certain non-vital equipment in the non-vital AC and DC buses.

There are three AC electrical power subsystems, each comprised of a primary ESF 4.16 kV bus and secondary 480 and 120 V buses, distribution panels, motor control centers and load centers. Each 4.16 kV ESF bus has two separate and independent offsite source of power as well as a dedicated onsite diesel generator (DG) source. Each 4.16 kV ESF bus is normally connected to the 500 kV offsite source. After a loss of this normal 500kV offsite power source to a 4.16 kV ESF bus, a transfer to the alternate 230 kV offsite source is accomplished by utilizing a time delayed bus undervoltage relay. If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ESF bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries. Additional description of this system may be found in the Bases for LCO 3.8.1, "AC Sources - Operating," and the Bases for LCO 3.8.4, "DC Sources - Operating."

The secondary 480 VAC electrical power distribution system for each bus includes the safety related motor control centers shown in Table B 3.8.9-1.

The 120 VAC vital buses are arranged in four buses and are normally powered from the inverters. The alternate power supply for the 120 VAC vital buses are Class 1E constant voltage source transformers powered from the same bus as the associated inverter, and its use is governed by LCO 3.8.7, "Inverters - Operating." Each constant voltage source transformer is powered from a Class 1E AC bus. In addition, each inverter can be powered from a bus other than its associated bus.

There are three independent 125 VDC electrical power distribution subsystems (one for each bus).

The list of all required distribution buses is presented in Table B 3.8.9-1.

(continued)

BASES

**APPLICABLE
SAFETY ANALYSES**

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 1), and in the FSAR, Chapter 15 (Ref. 2), assume ESF systems are OPERABLE. The Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

The OPERABILITY of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite power or all onsite AC electrical power; and
- b. A worst case single failure.

The distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The required power distribution subsystems listed in Table B 3.8.9-1 ensure the availability of Class 1E AC, DC, and 120 VAC vital bus electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems are required to be OPERABLE.

Maintaining the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

OPERABLE Class 1E AC electrical power distribution subsystems require the associated buses and motor control centers to be energized to their proper voltages. OPERABLE Class 1E DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger. OPERABLE 120 VAC vital bus electrical power distribution subsystems require

(continued)

BASES

LCO
(continued)

the associated buses to be energized to their proper voltage from the associated inverter via inverted DC voltage, inverter using internal AC source, or Class 1E constant voltage transformer.

In addition, tie breakers between redundant safety related Class 1E AC, DC, and 120 VAC vital bus power distribution subsystems, if they exist, must be open. This prevents any electrical malfunction in any power distribution subsystem from propagating to the redundant subsystem, that could cause the failure of a redundant subsystem and a loss of essential safety function(s). If any tie breakers are closed, the affected redundant electrical power distribution subsystems are considered inoperable. This applies to the onsite, safety related redundant electrical power distribution subsystems. It does not, however, preclude redundant Class 1E 4.16 kV buses from being powered from the same offsite circuit.

APPLICABILITY

The electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.10, "Distribution Systems - Shutdown."

ACTIONS

A.1

With one or more required Class 1E AC electrical power subsystems inoperable and a loss of function has not yet occurred, the remaining portions of the AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining portions of the power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the required Class 1E AC buses, load centers, and motor control centers must be restored to OPERABLE status within 8 hours.

Condition A worst scenario is one AC electrical power distribution subsystem without AC power (i.e., no offsite

(continued)

BASES

ACTION

A.1 (continued)

power to the 4160 V ESF bus and the associated DG inoperable). In this Condition, the unit is more vulnerable to a complete loss of AC power. It is, therefore, imperative that the unit operator's attention be focused on minimizing the potential for loss of power to the remaining AC electrical power distribution subsystems by stabilizing the unit, and on restoring power to the affected subsystem. The 8 hour time limit before requiring a unit shutdown in this Condition is acceptable because of:

- a. The potential for decreased safety if the unit operator's attention is diverted from the evaluations and actions necessary to restore power to the affected subsystem, to the actions associated with taking the unit to shutdown within this time limit; and
- b. The potential for an event in conjunction with a single failure of a redundant component in the other AC electrical power distribution subsystems with AC power.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 2 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the AC distribution system. At this time, a DC circuit could again become inoperable, and AC distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With one or more 120 VAC vital bus subsystems inoperable and a loss of function has not yet occurred, the remaining OPERABLE 120 VAC vital buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required AC vital bus

(continued)

BASES

ACTIONS

B.1 (continued)

subsystems must be powered from an alternate source within 2 hours by powering the bus from the associated inverter via inverted DC, inverter using internal AC source, or Class 1E constant voltage transformer. The required AC vital bus subsystems must then be re-powered by restoring it's associated inverter to OPERABLE status within 24 hours under LCO 3.8.7. ACTION A.1.

Condition B represents one 120 VAC vital bus without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptible power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining vital buses and restoring power to the affected 120 VAC vital bus subsystems.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate 120 VAC power. Taking exception to LCO 3.0.2 for components without adequate vital 120 VAC power, that would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

- a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate vital 120 VAC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected subsystem ; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the 120 VAC vital bus to OPERABLE status, the redundant capability afforded by the other OPERABLE 120 VAC vital buses, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to

(continued)

BASES

ACTIONS

B.1 (continued)

8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the 120 VAC vital bus distribution system. At this time, an AC bus could again become inoperable, and 120 VAC vital bus distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With one or more DC electrical power distribution subsystems inoperable and a loss of function has not yet occurred, the remaining portions of the DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining portion of the DC electrical power distribution subsystems could result in the minimum required ESF functions not being supported. Therefore, the DC buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

Condition C represents one or more DC electrical power distribution subsystems without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning for the affected bus(es). In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining DC electrical power distribution subsystems and restoring power to the affected subsystems.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

- a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;

(continued)

BASES

ACTIONS

C.1 (continued)

- b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected subsystem; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).

The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, an AC bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 8 hours. This could lead to a total of 10 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC bus could again become inoperable, and DC distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition C was entered. The 16 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

Condition E corresponds to required Class 1E AC, DC, or 120 VAC vital buses with inoperable distribution subsystems that result in a loss of safety function, adequate core cooling, containment OPERABILITY and other vital functions for DBA

(continued)

BASES

ACTIONS

E.1 (continued)

mitigation would be compromised, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE
REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the required Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC, DC, and 120 VAC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

Table B 3.8.9-1

Table on next page define the general features of the AC and DC Electrical Power Distribution System.

REFERENCES

1. FSAR, Chapter 6.
 2. FSAR, Chapter 15.
 3. Regulatory Guide 1.93, December 1974.
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BASES

Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

LCO 3.8.9 CONDITION A
4160 VAC and 480 VAC

VOLTAGE	BUS F MAJOR ESF LOADS (TRAIN A)	BUS G MAJOR ESF LOADS (TRAIN B)	BUS H MAJOR ESF LOADS (TRAIN A&B)
4160 VAC	ASW PP 1 AFW PP 3 CCP PP 1 CCW PP 1 SI PP 1 480 VAC BUS F	ASW PP 2 CS PP 1 RHR PP 1 CC PP 2 CCW PP 2 480 VAC BUS G	AFW PP 2 (B) CS PP 2 (A) RHR PP 2 (A) SI PP 2 (B) CCW PP 3 (A&B) 480 VAV BUS H
480 VAC *	CFCU 1 CFCU 2	CFCU 3 CFCU 5	CFCU 4 (A&B)

* Partial listing of loads

LCO 3.8.9 CONDITION B.
120 VAC

BUS 1 PY11 (21)** PY11A (21A)**	BUS 2 PY12 (22)**	BUS 3 PY13 (23)** PY13A (23A)**	BUS 4 PY14 (24)**
IY Powered by: 480 VAC BUS F/DC BUS 1 or TRY1 Powered by: 480 VAC BUS F or Backup 480 VAC BUS G	IY1 Powered by: 480 VAC BUS G/DC BUS 2 or TRY2 Powered by: 480 VAC BUS G or Backup 480 VAC BUS F	IY Powered by: 480 VAC BUS H/DC BUS 3 or TRY3 Powered by: 480 VAC BUS H or Backup 480 VAC BUS G	IY Powered by: 480 VAC BUS H/DC BUS 2 or TRY1 Powered by: 480 VAC BUS H or Backup 480 VAC BUS F

** Unit 2 in parentheses

LCO 3.8.9 CONDITION C
125 VDC

DC BUS 1 - Powered From:	DC BUS 2 - Powered From:	DC BUS 3 - Powered From:
Battery 1 and Batory Charger 11 (21)** or Battery Charger 121 (221)**	Battery 2 and Batory Charger 12 (22)** or Battery Changer 121 (221)**	Battery 3 and Batory Charger 131 (231)** or Battery Charger 132 (232)**

** Unit 2 in Parentheses

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND A description of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The Class 1E AC, DC, and 120 VAC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies ensures that:

- a. The unit can be maintained in the shutdown or refueling condition for extended periods;
- b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and
- c. Adequate power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The Class 1E AC, DC, and 120 VAC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. An OPERABLE AC subsystem shall consist of a 4kV vital bus powered from at least one energized offsite power source with the capability of being powered from an OPERABLE DG. The DG may be the DG associated with that bus or, with administrative

(continued)

BASES

LCO
(continued)

controls in place, a DG that can be cross-tied (via the startup cross-tie feeder breakers) to another bus. However, credit for this cross-tie capability cannot be taken credit for in those LCOs which specifically require an OPERABLE emergency power source. The latter ensures that the 4 kV bus will be immediately available after a LOOP without operator action. An OPERABLE DC subsystem consists of an OPERABLE DC bus (see B 3.8.5). An OPERABLE Class 1E 120 VAC subsystem consists of a vital 120 VAC bus that is powered by its OPERABLE inverter which is connected to an OPERABLE DC bus, or except as precluded by LCO 3.8.8, one that is powered from its associated vital 120 VAC regulating transformer that is selected to be powered from an OPERABLE AC vital bus. This ensures that the vital 120 VAC bus is capable of supplying either uninterruptable power from its associated inverter, or with administrative controls in place, from its vital 120 VAC regulating transformer after a brief time delay for the DG to load the bus following a LOOP. The 120 VAC regulating transformer must be capable of being energized without any operator action. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components—all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC, DC, and 120 VAC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of irradiated fuel assemblies, provide assurance that:

- a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;
- b. Systems needed to mitigate a fuel handling accident are available;
- c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

The AC, DC, and 120 VAC vital bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

(continued)

BASES

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, the movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5

Although redundant required features may require redundant subsystems of electrical power distribution systems to be OPERABLE, one OPERABLE distribution subsystem may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions).

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC, DC, and 120 VAC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal (RHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.4 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.8.10.1

This Surveillance verifies that the Class 1E AC, DC, and 120 VAC vital bus electrical power distribution subsystems are functioning properly, with all the buses energized. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. FSAR, Chapter 6.
 2. FSAR, Chapter 15.
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B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. The refueling boron concentration is sufficient to maintain shutdown margin (SDM) with the most adverse conditions of fuel assembly and control rod position allowed by plant procedures. The boron concentration that is maintained in Mode 6 is sufficient to maintain $k_{eff} \leq 0.95$ with the most reactive rod control assembly completely removed from its fuel assembly.

GDC 26 of 10 CFR 50, Appendix A, requires that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the principle system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with refueling grade borated water from the liquid hold up tanks or the refueling water storage tank.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level") to provide forced circulation cooling in the RCS and assist in maintaining the boron concentrations uniformity in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

(continued)

BASES

APPLICABLE
SAFETY ANALYSIS

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). It is based upon a maximum dilution flow of 300 g.p.m. and prompt identification and operation preclude the event from proceeding to a boron dilution accident. Prompt identification is assured through audible count rate instrumentation, a high count rate alarm and a high source range flux level alarm.

The RCS boron concentration satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7, "Control Bank Insertion Limits," ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

(continued)

BASES

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE
REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
 2. FSAR, Chapter 15, Section 15.2.4
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B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The installed source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range (source range drawer) covers six decades of neutron flux (10 to 1E+6 cps) with a $\pm 3\%$ instrument accuracy. The detectors also provide continuous visual indication in the control room and an audible alarm and count rate to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

APPLICABLE SAFETY ANALYSIS

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly. Prompt identification is required to assure sufficient time for operator action to preclude the event from proceeding to a Boron Dilution Accident. Prompt identification is assured through audible count rate indication, a high count rate alarm and a high source range flux level alarm in the control room.

The source range neutron flux monitors satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE, each monitor must provide visual indication and at least one of the two monitors must provide an audible alarm and count rate functions in the Control Room. Therefore, with no audible alarm and count rate functions from at least one monitor, both monitors are inoperable.

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are

(continued)

BASES (continued)

APPLICABILITY (continued) also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation" and LCO 3.3.9, "BDPS."

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. The exception given in A.1 for the process of latching/unlatching control rods and friction testing of control rods is provided to allow completion of head installation prior to replacing a failed source range detector. RCCA latching and friction testing is conducted with the reactor vessel upper internals in place, thereby preventing the lowering of a temporary source range detector into the region of the core. This NOTE allows control rod movement with only one source range in place. Friction testing involves fully withdrawing and reinserting each rod in turn, which could change core reactivity by as much as one percent for the most reactive rod. The increase in count rate would be one to two counts per second. The core coupling in this configuration would allow one source range detector to detect significant reactivity changes associated with control rod movement. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position or normal cooldown of a coolant volume for the purpose of system temperature control.

B.1

With no source range neutron flux monitor OPERABLE including no OPERABLE audible alarm and count rate functions, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor including no OPERABLE audible alarm and count rate functions is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

(continued)

BASES (continued)

SURVEILLANCE .
REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions. For core reload, the first CHANNEL CHECK for each channel may be performed using the first fuel assembly as a source, prior to unlatching it in the core.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The CHANNEL CALIBRATION also includes verification of the audible alarm and count rate functions on a simulated or actual boron dilution flux doubling signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.
 2. FSAR, Section 15.2.4.
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B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed by automatic means. Since any potential for containment pressurization yields very low levels, the 10CFR50, Appendix J leakage criteria and tests are not required. (Ref. 1)

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10CFR100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed for normal entry and exit.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be

(continued)

BASES

BACKGROUND
(continued)

restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System includes two subsystems. The normal subsystem includes a 48 inch purge penetration and a 48 inch exhaust penetration in which the flow path is limited to being open 200 hour or less per calendar year. The second subsystem, a pressure equalization system provides a single 12 inch supply and exhaust penetration. The three valves in the 12 inch pressure equalization penetration can be opened intermittently. Each of these system are qualified to closed automatically by the Engineered Safety Features Actuation System (ESFAS). Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchangers are necessary to conduct refueling operations. The normal 48 inch purge system is used for this purpose, and all four valves are closed by the ESFAS in accordance with LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

The pressure equalization system is disassembled and used in MODE 6 for other outage functions.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. The fuel transfer tube is open but closure is provided by an equivalent isolation of a water loop seal. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, ventilation barrier for the other containment penetrations during fuel movements (Ref. 1).

APPLICABLE
SAFETY ANALYSIS

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 2, consists of dropping a single irradiated fuel assembly. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values

Containment penetrations satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

(continued)

BASES

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure times specified in the FSAR can be achieved and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit.

LCO 3.9.4.c is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) Appropriate personnel are aware of the open status of the penetration flowpath during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment and 2) specified individuals are designated and readily available to isolate the flowpath in the event of a fuel handling accident.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE
REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates by inspection or administrative means that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal. The SR specifies that containment penetrations that are open under administrative controls are not required to meet the SR during the time the penetrations are open.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.4.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.6, the Containment Purge and Exhaust Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. Design Criteria Memorandum T-16, Containment Functions.
 2. FSAR , Section 15.4.5.
 3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.
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B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation - High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE
SAFETY ANALYSIS

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission product barrier.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in 10CFR50.36(c)(2)(ii) as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat

(continued)

BASES

LCO (continued) . removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

- a. Removal of decay heat;
- b. Mixing of borated coolant to minimize the possibility of criticality; and
- c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as valve testing, core mapping, or alterations in the vicinity of the reactor vessel hot leg nozzles. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

The LCO is also modified by a second Note that allows the required RHR Loop to be removed from service for up to 2 hours per 8 hour period to support surveillance leak rate testing of the RCS to RHR suction isolation valves, provided that no operations are permitted which might result in reduction of boron concentration. During this 2 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity and the RCS.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level \geq 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level $<$ 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Notes to the LCO.

(continued)

BASES

ACTIONS
(continued)

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. The suspension of any operation involving a reduction in reactor coolant boron concentration will reduce the likelihood of stratification of the boron concentration developing within the RCS.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading an irradiated fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level \geq 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

A.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE
REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate of 3000 gpm is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core prior to 57 hours of core subcriticality. The second part of this Surveillance serves the same function but with 57 hours or more of core subcriticality. The flow rate of 1300 gpm is determined by

(continued)

BASES

SURVEILLANCE
REQUIREMENTS -
(continued)

the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. Both of these flow rates are points of the same flow rate verses decay heat curves. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System (Ref. 2).

REFERENCES

1. FSAR , Section 5.5.7
 2. LAR 88-01, dated 4/21/88, submitted by "RHR System Flow Rate Reduction,"
DCL 88-067.
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B 3.9 REFUELING OPERATIONS

B 3.9.6 Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34, to provide mixing of borated coolant, and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchangers where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSIS

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, to prevent this challenge.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in 10CFR50.36(c)(2)(ii) as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

LCO

In MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, both RHR loops must be OPERABLE. Additionally, one loop of RHR must be in operation in order to provide:

- a. Removal of decay heat;
- b. Mixing of borated coolant to minimize the possibility of criticality; and
- c. Indication of reactor coolant temperature.

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. One

(continued)

BASES

LCO (continued) or both RHR pumps maybe aligned to the RWST to support filling the refueling cavity or for performance of required testing (Ref. 2).

APPLICABILITY Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level."

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no RHR loop is in operation, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS, because all of the unborated water sources are isolated.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing

(continued)

BASES

containment penetrations that are open to the outside atmosphere ensures that dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

SURVEILLANCE
REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate of more than 3000 gpm is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core prior to 57 hours subcritical. The second part of this Surveillance serves the same function but with 57 hours or more of core subcriticality and provides a reduced flow rate of 1300 gpm based upon a reduced decay heat load. Both of these flow rates are points of the same flow rate versus decay heat curves. The 1300 gpm limit also precludes exceeding the 1675 gpm upper flow limit to prevent vortexing and air entrainment of the RHR piping system. RHR pump vortexing (failure to meet pump suction requirements) during mid-loop operation may result in RHR pump failure and non-conservative RCS level indication. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room (Ref. 3).

SR 3.9.6.2

Verification that the required pump is OPERABLE ensures that an additional RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. FSAR, Section 5.5.7
 2. WOG Standard Technical Specification Change Traveler TSTF-21.
 3. LAR 88-01, dated 4/21/88, submitted by "RHR System Flow Rate Reduction,"
DCL 88-067.
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B 3.9 REFUELING OPERATIONS

B 3.9.7 Refueling Cavity Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies or performance of CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, or friction testing of individual control rods within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.

APPLICABLE SAFETY ANALYSIS

During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 100 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained well within allowable limits (Refs. 4, and 5.).

Refueling cavity water level satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits, as provided by the guidance of Reference 3.

(continued)

BASES

APPLICABILITY LCO 3.9.7 is applicable during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts, and when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.15, "Fuel Storage Pool Water Level."

ACTIONS A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving CORE ALTERATIONS or movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS SR 3.9.7.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

- 1.. Regulatory Guide 1.25, March 23, 1972.
2. FSAR, Section 15.4.5.
3. NUREG-0800, Section 15.7.4.
4. 10 CFR 100.10.
5. Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-828, Radiological Consequences of a Fuel Handling Accident, December 1971.
