



UNITED STATES  
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001  
March 7, 1994

Mr. Jim Eaton  
NUMARC  
1776 Eye Street, N.W.  
Suite 300  
Washington, D.C. 20006-2496

Dear Mr. Eaton:

Enclosed are preliminary copies of approved changes to the improved Standard Technical Specifications (STS) for confirmatory review by the Owners Groups. These changes have been through the STS change process established by the NRC, the Owners Groups, and NUMARC. The enclosed packages include approved changes from the following STS change packages, as listed in the Industry/NRC Improved Technical Specifications NUREG Change Packages Matrix:

- |          |         |                   |
|----------|---------|-------------------|
| •BWR-01  | •WOG-01 | •WOG-05           |
| •BWR-01A | •WOG-02 | •WOG-06           |
| •BWR-02  | •WOG-03 | •BWOG-01, C.1-C.7 |
| •BWR-03  | •WOG-04 |                   |

For each Owners Group there is a corresponding certification form for each change package. Please have the Owners Groups review the completed change pages for their associated NUREG and have the designated OG representative complete a certification form for each package.

During your review of the enclosed packages, you may notice that some minor format and grammatical corrections (spelling, capitalization, etc.) have been made that were not indicated in the change packages. We are confident that these corrections have not changed the intent or substance of the STS, but have improved the overall quality of the STS.

If any errors are identified that constitute differences from the originally approved changes, the designated OG representative should clearly note these on the certification form, where indicated. New changes, which were not part of the originally approved changes, should be submitted under a separate change package.

*DF03*  
*1/1*

003020

150088

9403180041 940307  
PDR NUREG  
1430 C PDR

*I+P-11-Guides + Manuals*  
*X-RD-10-1*  
*X-L-4-1-Technical Specifications*  
*X-O+M-7-NUMARC*

RETURN TO REGULATORY CENTRAL FILES

J. Eaton

- 2 -

We would appreciate the Owners Groups prompt review and certification of the enclosed changes so that we may make the approved changes publicly available as soon as possible. If you have any questions concerning these changes, please contact Nanette Gilles at (301) 504-1180 or Mary Lynn Reardon at (301) 504-1177.

Original Signed by:  
C. I. Grimes

Christopher I. Grimes, Chief  
Technical Specifications Branch  
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cc w/enclosure:  
D. Hoffman, EXCEL

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Distribution w/enclosure

Central Files  
PDR  
OTSB r/f  
NVGilles

Distribution w/o enclosure

IAReyes  
BKGrimes  
CIGrimes  
OTSB Staff

DOCUMENT NAME: G:GILLES\CHNG\_PKG.LTR

Concurrence

OTSB:DORS:NRR  
NVGilles *WJ*  
03/7/94

OTSB:DORS:NRR  
MLReardon *MLR*  
03/4/94

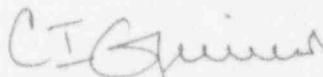
SC:OTSB:DORS:NRR  
FMReinhart *MR*  
03/7/94 *for*

C:OTSB:DORS:NRR  
CIGrimes *CG*  
03/7/94

J. Eaton

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# **BWOG**

**NUREG 1430**

## **AFFECTED BY CHANGE PACKAGES**

**BWR-02**

**WOG-01**

**WOG-02**

**WOG-04**

**WOG-05**

**WOG-06**

**BWOG-01**

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWR-02  
NUREG 1430

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-01  
NUREG 1430

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-02  
NUREG 1430

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-04  
NUREG 1430

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman



# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-05  
NUREG 1430

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-06  
NUREG 1430

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWOG-01  
NUREG 1430

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Technical Specification  
Subcommittee Chairman

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Technical Specification  
Subcommittee Chairman

1.0 USE AND APPLICATION

1.1 Definitions

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-----NOTE-----

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

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<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ALLOWABLE THERMAL POWER	ALLOWABLE THERMAL POWER shall be the maximum reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation.
AXIAL POWER IMBALANCE	AXIAL POWER IMBALANCE shall be the THERMAL POWER in the top half of the core, expressed as a percentage of RATED THERMAL POWER (RTP), minus the THERMAL POWER in the bottom half of the core, expressed as a percentage of RTP.
AXIAL POWER SHAPING RODS (APSRs)	APSRs shall be control components used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place cross calibration of the sensing elements and normal calibration of the remaining adjustable devices in the channel. Whenever a

(continued)

1.1 Definitions

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CHANNEL CALIBRATION  
(continued)

a sensing element is replaced, the next required inplace cross calibration consists of comparing the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

The CHANNEL CALIBRATION shall also include testing of safety related Reactor Protection System (RPS), Engineered Safety Feature Actuation System (ESFAS), and Emergency Feedwater Initiation and Control (EFIC) bypass functions for each channel affected by the bypass operation.

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display, and trip functions.
- b. Bistable channels (e.g., pressure switches and switch contacts)—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions.
- c. The ESFAS CHANNEL FUNCTIONAL TEST shall also include testing of ESFAS safety related bypass functions for each channel affected by bypass operation.

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(continued)

1.1 Definitions (continued)

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CONTROL RODS	CONTROL RODS shall be all full length safety and regulating rods that are used to shut down the reactor and control power level during maneuvering operations.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.9.1.6. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in [Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity"].
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > [15] minutes, making up at least 95% of the total noniodine activity in the coolant.

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(continued)

1.1 Definitions

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LEAKAGE  
(continued)

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

All LEAKAGE that is not identified LEAKAGE or controlled LEAKAGE;

c. Pressure Boundary LEAKAGE

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

MODE

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

NUCLEAR HEAT FLUX HOT CHANNEL FACTOR  $F_Q(Z)$

The NUCLEAR HEAT FLUX HOT CHANNEL FACTOR is the maximum local linear power density in the core divided by the core average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions.

NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ( $F_{\Delta H}^N$ )

The NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR is the ratio of the integral of linear power along the fuel rod on which minimum departure from nucleate boiling ratio occurs, to the average fuel rod power.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant

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1.1 Definitions

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OPERABLE - OPERABILITY  
(continued)

instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS

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

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

PRESSURE AND  
TEMPERATURE LIMITS  
REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.9.1.7. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

QUADRANT POWER TILT  
(QPT)

QPT is defined by the following equation and is expressed as a percentage.

$$QPT = 100 \left( \frac{\text{Power in any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1 \right)$$

RATED THERMAL POWER  
(RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of [2544] MWt.

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(continued)



1.1 Definitions (continued)

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REACTOR PROTECTION  
SYSTEM (RPS) RESPONSE  
TIME

The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power is interrupted at the CONTROL ROD drive trip breakers. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

SHUTDOWN MARGIN (SDM)

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

- a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn;
- b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the [nominal zero power design level]; and
- c. There is no change in APSR position.

With any CONTROL RODS not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during  $n$  Surveillance Frequency intervals, where  $n$  is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER

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

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## 1.0 USE AND APPLICATION

### 1.2 Logical Connectors

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**PURPOSE**                    The purpose of this section is to explain the meaning of logical connectors.

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

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

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

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**EXAMPLES**                    The following examples illustrate the use of logical connectors.

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## 1.0 USE AND APPLICATION

### 1.3 Completion Times

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

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

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

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

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

(continued)

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1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-2

ACTIONS

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

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

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

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

(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-4

ACTIONS

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

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

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

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

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3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

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

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

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

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

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

Exceptions to this Specification are stated in the individual Specifications.

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

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

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LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This

(continued)

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### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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

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

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

If a Required Action requires performance of a Surveillance or its Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

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

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

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3.0 SR APPLICABILITY

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SR 3.0.3            When the Surveillance is performed within the delay period  
(continued)           and the Surveillance is not met, the LCO must immediately be  
                         declared not met, and the applicable Condition(s) must be  
                         entered.

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SR 3.0.4            Entry into a MODE or other specified condition in the  
                         Applicability of an LCC shall not be made unless the LCO's  
                         Surveillances have been met within their specified  
                         Frequency. This provision shall not prevent passage through  
                         or to MODES or other specified conditions in compliance with  
                         Required Actions.

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two [required] containment cooling trains inoperable.	D.1 Restore one [required] containment cooling train to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 5.	6 hours  36 hours
F. Two containment spray trains inoperable.  <u>OR</u> Any combination of three or more trains inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.6.1 Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

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5.0 ADMINISTRATIVE CONTROLS

5.6 Technical Specifications (TS) Bases Control

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- 5.6.1 Changes to the Bases of the TS shall be made under appropriate administrative controls and reviewed according to Specification 5.5.1.
- 5.6.2 Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
- a. A change in the TS incorporated in the license; or
  - b. A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- 5.6.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- 5.6.4 Proposed changes that meet the criteria of Specification 5.6.2.a or Specification 5.6.2.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71.
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## 5.0 ADMINISTRATIVE CONTROLS

### 5.7 Procedures, Programs, and Manuals

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#### 5.7.1 Procedures

##### 5.7.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
- b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in [Generic Letter 82-33];
- c. Security plan implementation;
- d. Emergency plan implementation;
- e. Quality assurance for effluent and environmental monitoring;
- f. Fire Protection Program implementation; and
- g. All programs specified in Specification 5.7.2.

##### 5.7.1.2 Review and Approval

Each procedure of Specification 5.7.1.1, and changes thereto, shall be reviewed in accordance with Specification 5.5.1, approved by the [Plant Superintendent] or his designee in accordance with approved administrative procedures prior to implementation, except as specified in Specification 5.7.1.3, and reviewed periodically as set forth in administrative procedures.

##### 5.7.1.3 Temporary Changes

Temporary changes to procedures of Specification 5.7.1.1 may be made provided:

- a. The intent of the existing procedure is not altered;
- b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator license on the unit affected; and

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.2 Process Control Program (PCP) (continued)

- b. Shall be effective after review and acceptance by the [review method of Specification 5.5.1] and the approval of the [Plant Superintendent].

5.7.2.3 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program; and
- b. The ODCM shall also contain the Radioactive Effluent Controls and Radiological Environmental Monitoring programs required by Specification 5.7.2.7 and Specification 5.7.2.8, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification [5.9.1.3] and Specification [5.9.1.4].

Licensee initiated changes to the ODCM:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
  - 1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and
  - 2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
- b. Shall become effective after review and acceptance by the [review method of Specification 5.5.1] and the approval of the [Plant Superintendent]; and

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2 Programs and Manuals (continued)

5.7.2.14 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.7.2.15 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in [Regulatory Guide ], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1].

- a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < [0.05%] when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [± 10%].

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.15 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flowrate
[ ]	[ ]

- b. Demonstrate for each of the ESF systems that an in-place test of the charcoal adsorber shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [ $\pm 10\%$ ].

ESF Ventilation System	Flowrate
[ ]	[ ]

- c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of  $\leq$  [30°C] and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
[ ]	[ ]	[ ]

Reviewer's Note: Allowable penetration = [100% - methyl iodide efficiency for charcoal credited in staff safety evaluation]/ (safety factor).

Safety factor = [5] for systems with heaters.  
= [7] for systems without heaters.

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.17 Diesel Fuel Oil Testing Program (continued)

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
  - 1. an API gravity or an absolute specific gravity within limits,
  - 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
  - 3. a clear and bright appearance with proper color;
- b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and
- c. Total particulate concentration of the fuel oil is  $\leq 10$  mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.7.2.18 Fire Protection Program

This program provides controls to ensure that appropriate fire protection measures are maintained to protect the plant from fire and to ensure the capability to achieve and maintain safe shutdown in the event of a fire is maintained.

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BASES

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APPLICABILITY  
(continued)      In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

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SAFETY LIMIT  
VIOLATIONS      The following SL violation responses are applicable to the reactor core SLs.

2.2.1 and 2.2.2

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the requirement to go to MODE 3 places the plant in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

2.2.5

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3).

2.2.6

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

2.2.7

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 4). A copy of the report shall also be submitted to the senior

(continued)



BASES

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SAFETY LIMIT  
VIOLATIONS

2.2.7 (continued)

management of the nuclear plant, and the utility Vice President - Nuclear Operations, and the [offsite reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.8

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988.
  2. FSAR, Section [ ].
  3. 10 CFR 50.72.
  4. 10 CFR 50.73.
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BASES

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SAFETY LIMIT  
VIOLATIONS  
(continued)

2.2.6

If the RCS pressure SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

2.2.7

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC, in accordance with 10 CFR 50.73 (Ref. 9). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations and the [offsite reviewers specified in Specification 5.2.2] ["Offsite Review and Audit"].

2.2.8

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

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28, 1988.
2. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
4. BAW-10043, May 1972.
5. FSAR, Section [14].
6. ASME USAS B31.1, Standard Code for Pressure Piping, 1967.

(continued)

BASES

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REFERENCES  
(continued)

7. 10 CFR 100.
  8. 10 CFR 50.72.
  9. 10 CFR 50.73.
- 
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BASES

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SR 3.0.2  
(continued)

Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

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SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most

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BASES

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SR 3.0.3  
(continued)

probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance. SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. This Specification applies to

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BASES

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SR 3.0.4  
(continued)

changes in MODES or other specified conditions in the Applicability associated with unit shutdown as well as startup.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

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## B 3.1 REACTIVITY CONTROL SYSTEMS

### B 3.1.5 Safety Rod Insertion Limit

#### BASES

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#### BACKGROUND

The insertion limits of the safety and regulating rods are initial condition assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected rod worth, and initial reactivity insertion rate.

The applicable criteria for the reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," and GDC 26, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on safety rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved.

The regulating groups are used for precise reactivity control of the reactor. The positions of the regulating groups are normally automatically controlled by the automatic control system, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The regulating groups must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature and fuel burnup.

The safety groups can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The safety groups are controlled manually by the control room operator. During normal full power operation, the safety groups are fully withdrawn. The safety groups must be completely withdrawn from the core prior to withdrawing any regulating groups during an approach to criticality. The safety groups

(continued)

BASES

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ACTIONS  
(continued)

C.2.2

The SDM and ejected rod worth limit can also be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion limits in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the plant systems. Operation for up to 2 hours more in the restricted region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution.

D.1

If the regulating rods cannot be restored to within the acceptable operating limits for the original THERMAL POWER, or if the power reduction cannot be completed within the required Completion Time, then the reactor is placed in MODE 3, in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the amount of time required to reach MODE 3 from RTP without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.1.1

This Surveillance ensures that the sequence and overlap limits are not violated. A Surveillance Frequency of 12 hours or 4 hours, depending on whether the CONTROL ROD drive sequence alarm is OPERABLE or not, is acceptable because little rod motion occurs in 4 hours due to fuel burnup, and the probability of a deviation occurring simultaneously with an inoperable sequence monitor in this relatively short time frame is low. Also, the Frequency

(continued)



BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.1.1 (continued)

takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.2

With an OPERABLE regulating rod insertion limit alarm, verification of the regulating rod insertion limits as specified in the COLR at a Frequency of 12 hours is sufficient to ensure the OPERABILITY of the regulating rod insertion limit alarm and to detect regulating rod banks that may be approaching the group insertion limits, because little rod motion due to fuel burnup occurs in 12 hours. If the insertion limit alarm becomes inoperable, verification of the regulating rod group position at a Frequency of 4 hours is sufficient to detect whether the regulating rod groups may be approaching or exceeding their group insertion limits, although more frequent surveillance is prudent if the regulating rod insertion limit alarm is not OPERABLE. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.3

Prior to achieving criticality, an estimated critical position for the CONTROL RODS is determined. Verification that SDM meets the minimum requirements ensures that sufficient SDM capability exists with the CONTROL RODS at the estimated critical position if it is necessary to shut down or trip the reactor after criticality. The Frequency of 4 hours prior to criticality provides sufficient time to verify SDM capability and establish the estimated critical position.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
2. 10 CFR 50.46.

(continued)

BASES

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REFERENCES  
(continued)

3. FSAR, Section [ ].
  4. FSAR, Section [ ].
  5. FSAR, Section [ ].
  6. FSAR, Section [ ].
  7. FSAR, Section [ ].
  8. FSAR, Section [ ].
- 
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BASES

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ACTIONS  
(continued)

the APSR withdrawal. If this occurs, the APSRs must be restored to their normal inserted position. Conversely, after the fuel cycle burnup for the APSR withdrawal occurs, the APSRs may not be reinserted for the remainder of the fuel cycle. These restrictions apply to ensure the axial burnup distribution that accumulates in the fuel will be consistent with the expected (as designed) distribution.

A.1

For verification that the core parameters  $F_0(Z)$  and  $F_{\Delta H}^N$  are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Successful verification that  $F_0(Z)$  and  $F_{\Delta H}^N$  are within their limits ensures that operation with the APSRs inserted or withdrawn in violation of the times specified in the COLR do not violate either the ECCS or DNB criteria (Ref. 4). The required Completion Time of 2 hours is reasonable to allow the operator to obtain a power distribution map and to verify the power peaking factors. Repeating SR 3.2.5.1 every 2 hours is reasonable to ensure that continued verification of the power peaking factors is obtained as core conditions (primarily the regulating rod insertion and induced xenon redistribution) change.

A.2

Indefinite operation with the APSRs inserted or withdrawn in violation of the times specified in the COLR is not prudent. Even if power peaking monitoring per Required Action A.1 is continued, the abnormal APSR insertion or withdrawal may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may affect the long term fuel depletion pattern. Therefore, power peaking monitoring is allowed for up to 24 hours. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the APSR limit out of specification. In addition, it precludes long term depletion with the APSRs in positions that have not been analyzed, thereby limiting the potential for an adverse xenon redistribution. This time limit also ensures that the intended burnup distribution is maintained, and allows the operator sufficient time to reposition the APSRs to correct their positions.

(continued)

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BASES

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ACTIONS

A.2 (continued)

Because the APSRs are not operated by the automatic control system, manual action by the operator is required to restore the APSRs to the positions specified in the COLR.

B.1

If the APSRs cannot be restored to their intended positions within the required Completion Time of 24 hours, the reactor must be placed in MODE 3, in which this LCO does not apply. This action ensures that the fuel does not continue to be depleted in an unintended burnup distribution. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 3 from RTP in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.2.1

Fuel cycle designs that allow APSR withdrawal near EOC do not permit reinsertion of APSRs after the time of withdrawal. When the plant computer is OPERABLE, the operator will receive a computer alarm if the APSRs insert after that time in core life when the APSR withdrawal occurs. Verification that the APSRs are within their insertion limits at a 12 hour Frequency is sufficient to ensure that the APSR insertion limits are preserved and the computer alarm remains OPERABLE. The 12 hour Frequency required for performing this verification is sufficient because APSRs are positioned by manual control and are normally moved infrequently. The probability of a deviation occurring simultaneously with an inoperable computer alarm is low in this relatively short time frame. Also, the Frequency takes into account other information available in the control room for monitoring the axial power distribution in the reactor core.

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(continued)

BASES

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ACTIONS  
(continued)

D.1

With two of the required containment cooling trains inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. The components in this degraded condition (both spray trains are OPERABLE or else Condition E is entered) provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

E.1 and E.2

If the Required Actions and associated Completion Times of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With two containment spray trains or any combination of three or more containment spray and containment cooling trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in

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BASES

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ACTIONS

A.3 (continued)

Event that must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.1.1

This SR ensures the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is therefore a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.3.1 (continued)

radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.15, "RB Purge Isolation-High Radiation," the isolation instrumentation requires a CHANNEL CHECK every 12 hours and a CHANNEL FUNCTIONAL TEST every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

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REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.
  2. FSAR, Section [ ].
  3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.
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# **CEOG**

**NUREG 1432**

**AFFECTED BY  
CHANGE PACKAGES**

**BWR-02**

**WOG-01**

**WOG-04**

**WOG-06**

**BWOG-01**



## STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWR-02  
NUREG 1432

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-01  
NUREG 1432

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-04  
NUREG 1432

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-06  
NUREG 1432

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

## STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWOG-01  
NUREG 1432

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

1.1 Definitions

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CHANNEL CALIBRATION  
(continued)

the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place cross calibration of the sensing elements and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required in-place cross calibration consists of comparing the other sensing elements with the recently installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display and trip functions;
- b. Bistable channels (e.g., pressure switches and switch contacts)—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions; or

(continued)

1.1 Definitions

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CHANNEL FUNCTIONAL TEST  
(continued)

- c. Digital computer channels—the use of diagnostic programs to test digital computer hardware and the injection of simulated process data into the channel to verify OPERABILITY, including alarm and trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components [excluding control element assemblies (CEAs) withdrawn into the upper guide structure] affecting reactivity, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS  
REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.9.1.6. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

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

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(continued)

1.1 Definitions

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LEAKAGE  
(continued)

b. Unidentified LEAKAGE

All LEAKAGE that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

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

MODE

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

OPERABLE - OPERABILITY

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

PHYSICS TESTS

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

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

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(continued)



1.1 Definitions

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SHUTDOWN MARGIN (SDM) (continued)	With any CEAs not capable of being fully inserted, the reactivity worth of these CEAs must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during $n$ Surveillance Frequency intervals, where $n$ is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

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## 1.0 USE AND APPLICATION

### 1.2 Logical Connectors

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**PURPOSE**            The purpose of this section is to explain the meaning of logical connectors.

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

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

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

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**EXAMPLES**            The following examples illustrate the use of logical connectors.

(continued)

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## 1.0 USE AND APPLICATION

### 1.3 Completion Times

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

---

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

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

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

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

(continued)

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1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLES 1.3-2

ACTIONS

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

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

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

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

(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-4

ACTIONS

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

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

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

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

(continued)

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

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

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

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

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

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

Exceptions to this Specification are stated in the individual Specifications.

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

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

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LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This

(continued)

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### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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

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

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

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

Exceptions to this Specification are stated in the individual Specifications.

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

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

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be

(continued)

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3.0 SR APPLICABILITY

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SR 3.0.3            declared not met, and the applicable Condition(s) must be  
(continued)        entered.

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SR 3.0.4            Entry into a MODE or other specified condition in the  
Applicability of an LCO shall not be made unless the LCO's  
Surveillances have been met within their specified  
Frequency. This provision shall not prevent passage through  
or to MODES or other specified conditions in compliance with  
Required Actions.

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Containment Spray and Cooling Systems (Atmospheric and Dual)  
3.6.6A

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two containment cooling trains inoperable.	D.1 Restore one containment cooling train to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3.	6 hours
	<u>AND</u> E.2 Be in MODE 5.	36 hours
F. Two containment spray trains inoperable.  <u>OR</u>  Any combination of three or more trains inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.6A.1 Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

(continued)

Containment Spray and Cooling Systems (Atmospheric and Dual)  
3.6.6B

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. One containment spray train and one containment cooling train inoperable.	D.1    Restore containment spray train to OPERABLE status.	72 hours
	<u>OR</u>	
	D.2    Restore containment cooling train to OPERABLE status.	72 hours
E. Two containment cooling trains inoperable.	E.1    Restore one containment cooling train to OPERABLE status.	72 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1    Be in MODE 3.	6 hours
	<u>AND</u>	
	F.2    Be in MODE 5.	36 hours
G. Any combination of three or more trains inoperable.	G.1    Enter LCO 3.0.3.	Immediately

5.0 ADMINISTRATIVE CONTROLS

5.6 Technical Specifications (TS) Bases Control

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- 5.6.1 Changes to the Bases of the TS shall be made under appropriate administrative controls and reviewed according to Specification 5.5.1.
- 5.6.2 Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
- a. A change in the TS incorporated in the license; or
  - b. A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- 5.6.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- 5.6.4 Proposed changes that meet the criteria of Specification 5.6.2.a or Specification 5.6.2.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71.
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5.0 ADMINISTRATIVE CONTROLS

5.7 Procedures, Programs, and Manuals

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5.7.1 Procedures

5.7.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
- b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in [Generic Letter 82-33];
- c. Security plan implementation;
- d. Emergency plan implementation;
- e. Quality assurance for effluent and environmental monitoring;
- f. Fire Protection Program implementation; and
- g. All programs specified in Specification 5.7.2.
- h. Modification of core protection calculator (CPC) addressable constants. These procedures shall include provisions to ensure that sufficient margin is maintained in CPC type I addressable constants to avoid excessive operator interaction with CPCs during reactor operation.

Modifications to the CPC software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with the most recent version of "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, which has been determined to be applicable to the facility. Additions or deletions to CPC addressable constants or changes to addressable constant software limit values shall not be implemented without prior NRC approval.

5.7.1.2 Review and Approval

Each procedure of Specification 5.7.1.1, and changes thereto, shall be reviewed in accordance with Specification 5.5.1, approved

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.1.2 Review and Approval (continued)

by the [Plant Superintendent] or his designee in accordance with approved administrative procedures prior to implementation, except as specified in Specification 5.7.1.3, and reviewed periodically as set forth in administrative procedures.

5.7.1.3 Temporary Changes

Temporary changes to procedures of Specification 5.7.1.1 may be made provided:

- a. The intent of the existing procedure is not altered;
- b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator license on the unit affected; and
- c. The change is documented and reviewed in accordance with Specification 5.5.1 and approved by the [Plant Superintendent] or his designee in accordance with approved administrative procedures within 14 days of implementation.

5.7.2 Programs and Manuals

The following programs shall be established, implemented, and maintained.

5.7.2.1 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

5.7.2.2 Process Control Program (PCP)

The PCP shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes will be accomplished to ensure compliance with 10 CFR 20, 10 CFR 61, and 10 CFR 71; state regulations; burial ground requirements; and other requirements governing the disposal of solid radioactive waste.

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.2 Process Control Program (PCP) (continued)

Licensee initiated changes to the PCP:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
  1. Sufficient information to support the change(s) and appropriate analyses or evaluations justifying the change(s), and
  2. A determination that the change(s) maintain the overall conformance of the solidified waste product to the existing requirements of Federal, State, or other applicable regulations.
- b. Shall be effective after review and acceptance by the [review method of Specification 5.5.1] and the approval of the [Plant Superintendent].

5.7.2.3 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program; and
- b. The ODCM shall also contain the Radioactive Effluent Controls and Radiological Environmental Monitoring programs required by Specification 5.7.2.7 and Specification 5.7.2.8, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification [5.9.1.3] and Specification [5.9.1.4].

Licensee initiated changes to the ODCM:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
  1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

(continued)

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5.7 Procedures, Programs, and Manuals

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5.7.2.13 Steam Generator (SG) Tube Surveillance Program (continued)

- c. SG tube plugging/repair limits. These limits shall be [40]% of the nominal tube wall thickness consistent with [Regulatory Guide 1.83, Revision [ ], date].
- d. Specific definitions and limits for SG tube inservice inspection acceptance criteria consistent with [Regulatory Guide 1.83, Revision [ ], date].

The content and frequency of written reports shall be in accordance with Specification 5.9.2.

The provisions of SR 3.0.2 are applicable to SG Tube Surveillance Program inspection frequencies except those established by Category C-3 inspection results.

[Key elements to be discussed and provided.]

5.7.2.14 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2 Programs and Manuals (continued)

5.7.2.15 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in [Regulatory Guide ], and in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [ $\pm 10\%$ ].

- a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass  $< [0.05]\%$  when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989, at the system flowrate specified as follows [ $\pm 10\%$ ]:

ESF Ventilation System

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Flowrate

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- b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass  $< [0.05]\%$  when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified as follows [ $\pm 10\%$ ]:

ESF Ventilation System

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Flowrate

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- c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of  $\leq [30^{\circ}\text{C}]$  and greater than or equal to the relative humidity specified as follows:

(continued)



5.7 Procedures, Programs, and Manuals

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5.7.2.16 Explosive Gas and Storage Tank Radioactivity Monitoring Program  
(continued)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.7.2.17 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
  1. An API gravity or an absolute specific gravity within limits,
  2. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
  3. A clear and bright appearance with proper color;
- b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and
- c. Total particulate concentration of the fuel oil is  $\leq 10$  mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.7.2.18 Fire Protection Program

This program provides controls to ensure that appropriate fire protection measures are maintained to protect the plant from fire and to ensure the capability to achieve and maintain safe shutdown in the event of a fire is maintained.

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BASES

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SAFETY LIMIT  
VIOLATIONS  
(continued)

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

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

2.2.3

If SL 2.1.1 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3).

2.2.4

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

2.2.5

If SL 2.1.1 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 4). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations and the [offsite reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.6

If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and

(continued)

BASES

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SAFETY LIMIT  
VIOLATIONS

2.2.6 (continued)

actions are completed before the unit begins its restart to normal operation.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988.
  2. FSAR, Section [ ].
  3. 10 CFR 50.72.
  4. 10 CFR 50.73.
- 
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BASES

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SAFETY LIMIT  
VIOLATIONS  
(continued)

2.2.3

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

2.2.4

If the RCS pressure SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and to assess the condition of the unit before reporting to senior management.

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations and the [offsite reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.6

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

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.

(continued)

BASES

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REFERENCES  
(continued)

4. 10 CFR 100.
  5. FSAR, Section [ ].
  6. ASME, USAS B31.1, Standard Code for Pressure Piping, 1967. [ ]
  7. 10 CFR 50.72.
  8. 10 CFR 50.73.
- 
-

BASES

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APPLICABILITY (continued)      function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

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SAFETY LIMIT VIOLATIONS      The following violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

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

2.2.3

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3).

2.2.4

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.5

If SL 2.1.1.1 or SL 2.1.1.2 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 4). A copy of the

(continued)

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BASES

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SAFETY LIMIT  
VIOLATION

2.2.5 (continued)

report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations and the [offsite reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.6

If SL 2.1.1.1 or SL 2.1.1.2 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988.
  2. FSAR, Section [ ].
  3. 10 CFR 50.72.
  4. 10 CFR 50.73.
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BASES

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SAFETY LIMIT  
VIOLATIONS  
(continued)

2.2.3

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

2.2.4

If the RCS pressure SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and to assess the condition of the unit before reporting to the senior management.

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations and the [offsite reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.6

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

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.

(continued)



BASES

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REFERENCES  
(continued)

4. 10 CFR 100.
  5. FSAR, Section [ ].
  6. ASME, USAS B31.1, Standard Code for Pressure Piping, 1967. [ ]
  7. 10 CFR 50.72.
  8. 10 CFR 50.73.
- 
-

BASES

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SR 3.0.3  
(continued)

been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay

(continued)

BASES

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SR 3.0.3  
(continued)

period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified Condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. This Specification applies to changes in MODES or other specified conditions in the Applicability associated with unit shutdown as well as startup.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

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BASES

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ACTIONS

C.1 (continued)

The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

D.1

With two required containment cooling trains inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System, the iodine removal function of the Containment Spray System, and the low probability of a DBA occurring during this period.

E.1 and E.2

If the Required Actions and associated Completion Times of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With two containment spray trains or any combination of three or more Containment Spray System and Containment Cooling System trains inoperable, the unit is in a condition

(continued)

BASES

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ACTIONS

F.1 (continued)

outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

---

SURVEILLANCE  
REQUIREMENTS

SR 3.6.6A.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verifying, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.6A.2

Operating each containment cooling train fan unit for  $\geq 15$  minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected and corrective action taken. The 31 day Frequency of this SR was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of a significant degradation of the containment cooling train occurring between surveillances and has been shown to be acceptable through operating experience.

SR 3.6.6A.3

Verifying a service water flow rate of  $\geq [2000]$  gpm to each cooling unit provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 2). Also considered in selecting this Frequency were the known reliability of the Cooling Water System, the two train

(continued)

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BASES

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ACTIONS  
(continued)

E.1

With two containment cooling trains inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing greater than 100% of the heat removal needs after an accident. The 72 hour Completion Time was developed based on the same reasons as those for Required Action C.1.

F.1 and F.2

If any of the Required Actions and associated Completion Times of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

G.1

With any combination of three or more Containment Spray System and Containment Cooling System trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

---

SURVEILLANCE  
REQUIREMENTS

SR 3.6.6B.1

Verifying the correct alignment for manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR also does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct positions prior to being secured. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that

(continued)

**BWR/4**

**NUREG 1433**

**AFFECTED BY  
CHANGE PACKAGES**

**BWR-01**

**BWR-01A**

**BWR-02**

**BWR-03**

**WOG-01**

**WOG-06**

**BWOG-01**

## STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWR-01  
NUREG 1433

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

---

Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

---

Technical Specification  
Subcommittee Chairman



# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWR-01A  
NUREG 1433

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Subcommittee Chairman

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWR-02  
NUREG 1433

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Technical Specification  
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## STS CHANGE CERTIFICATION FORM

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NUREG 1433

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Technical Specification  
Subcommittee Chairman

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CHANGE PACKAGE WOG-01  
NUREG 1433

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Subcommittee Chairman

## STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-06  
NUREG 1433

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Subcommittee Chairman

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWOG-01  
NUREG 1433

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

1.0 USE AND APPLICATION

1.1 Definitions

-----NOTE-----

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

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AVERAGE PLANAR EXPOSURE	The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.
AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	The APLHGR shall be applicable to a specific planar height and is equal to the sum of the [LHGRs] [heat generation rate per unit length of fuel rod] for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle [at the height].
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place cross calibration of the sensing elements and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required in-place cross calibration consists of comparing the other sensing elements with the recently

(continued)

## 1.1 Definitions

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### CHANNEL CALIBRATION (continued)

installed sensing element. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

### CHANNEL CHECK

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

### CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips.
- b. Bistable channels (e.g., pressure switches and switch contacts)—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

### CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity, within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement) is not considered a CORE ALTERATION.

(continued)



1.1 Definitions

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CORE ALTERATION  
(continued)

In addition, control rod movement with other than the normal control rod drive is not considered a CORE ALTERATION provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS  
REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.9.1.6. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

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

$\bar{E}$  - AVERAGE  
DISINTEGRATION ENERGY

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

EMERGENCY CORE COOLING  
SYSTEM (ECCS) RESPONSE  
TIME

The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their

(continued)

## 1.1 Definitions

EMERGENCY CORE COOLING  
SYSTEM (ECCS) RESPONSE  
TIME

(continued)

required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

END OF CYCLE  
RECIRCULATION PUMP TRIP  
(EOC RPT) SYSTEM RESPONSE  
TIME

The EOC RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by [the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint] to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured, [except for the breaker arc suppression time, which is not measured but is validated to conform to the manufacturer's design value].

ISOLATION SYSTEM  
RESPONSE TIME

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

## LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or
2. LEAKAGE into the drywell atmosphere from sources that are both specifically located

(continued)

## 1.1 Definitions

LEAKAGE  
(continued)

and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

b. Unidentified LEAKAGE

All LEAKAGE into the drywell that is not identified LEAKAGE;

c. Total LEAKAGE

Sum of the identified and unidentified LEAKAGE;

d. Pressure Boundary LEAKAGE

LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.

LINEAR HEAT GENERATION  
RATE (LHGR)

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL  
TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

MAXIMUM FRACTION  
OF LIMITING  
POWER DENSITY (MFLPD)

The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

MINIMUM CRITICAL POWER  
RATIO (MCPR)

The MCPR shall be the smallest critical power ratio (CPR) that exists in the core [for each class of fuel]. The CPR is that power in the assembly that is calculated by application of the

(continued)

1.1 Definitions

---

MINIMUM CRITICAL POWER RATIO (MCPR) (continued)	appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:  a. Described in Chapter [14, Initial Test Program] of the FSAR;  b. Authorized under the provisions of 10 CFR 50.59; or  c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.9.1.7. Plant operation within these operating limits is addressed in LCO 3.4.10, "RCS Pressure and Temperature (P/T) Limits."

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(continued)

## 1.1 Definitions (continued)

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RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of [2436] MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	<p>SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:</p> <ol style="list-style-type: none"><li>The reactor is xenon free;</li><li>The moderator temperature is 68°F; and</li><li>All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn.</li></ol> <p>With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.</p>
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during $n$ Surveillance Frequency intervals, where $n$ is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

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(continued)

1.1 Definitions (continued)

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TURBINE BYPASS SYSTEM  
RESPONSE TIME

The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:

- a. The time for initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and
- b. The time for initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

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## 1.0 USE AND APPLICATION

### 1.2 Logical Connectors

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**PURPOSE**                    The purpose of this section is to explain the meaning of logical connectors.

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

---

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

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

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**EXAMPLES**                    The following examples illustrate the use of logical connectors.

(continued)

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## 1.0 USE AND APPLICATION

### 1.3 Completion Times

---

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

---

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

---

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

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

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

(continued)

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### 1.3 Completion Times

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DESCRIPTION  
(continued)

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

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

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

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

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

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

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(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)

### 1.3 Completion Times

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#### EXAMPLES

#### EXAMPLE 1.3-3 (continued)

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

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

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

(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

---

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

---

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

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

---

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

- a. MODE 2 within 7 hours;
- b. MODE 3 within 13 hours; and
- c. MODE 4 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

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

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

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LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This

(continued)

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### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

---

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

---

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

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

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

Exceptions to this Specification are stated in the individual Specifications.

---

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

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

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be

(continued)

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3.0 SR APPLICABILITY

---

SR 3.0.3 declared not met, and the applicable Condition(s) must be  
(continued) entered.

---

SR 3.0.4 Entry into a MODE or other specified condition in the  
Applicability of an LCO shall not be made unless the LCO's  
Surveillances have been met within their specified  
Frequency. This provision shall not prevent passage through  
or to MODES or other specified conditions in compliance with  
Required Actions.

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.4.1 -----NOTE-----                      Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 Item b or c requirements.                      -----</p> <p>Verify MFLPD is within limits.</p>	<p>Once within                      12 hours after                      ≥ 25% RTP</p> <p><u>AND</u></p> <p>24 hours                      thereafter</p>
<p>SR 3.2.4.2 -----NOTE-----                      Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 Item a requirements.                      -----</p> <p>Verify APRM setpoints or gains are adjusted for the calculated MFLPD.</p>	<p>12 hours</p>



Table 3.3.1.1-1 (page 1 of 3)  
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors	2	[3]	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ [120/125] divisions of full scale
a. Neutron Flux - High	5(a)	[3]	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ [120/125] divisions of full scale
b. Inop	2	[3]	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	[3]	H	SR 3.3.1.1.5 SR 3.3.2.2.15	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ [20]% RTP
b. Flow Biased Simulated Thermal Power - High	1	[2]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SF 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ [0.58 W + 62]% RTP and ≤ [115.5]% RTP(b)

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) [0.58 W + 62% - 0.58 ΔW]RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Table 3.3.1.1-1 (page 2 of 3)  
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux - High	1	[2]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ [120]% RTP
d. Downscale	1	[2]	F	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.15	≥ [3]% RTP
e. Inop	1,2	[2]	G	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.15	NA
3. Reactor Vessel Steam Dome Pressure - High	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.9 [SR 3.3.1.1.10] SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ [1054] psig
4. Reactor Vessel Water Level - Low, Level 3	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.9 [SR 3.3.1.1.10] SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ [10] inches
5. Main Steam Isolation Valve - Closure	1	[8]	F	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ [10]% closed
6. Drywell Pressure - High	1,2	[2]	G	SR 3.3.1.1.1 SR 3.3.1.1.9 [SR 3.3.1.1.10] SR 3.3.1.1.13 SR 3.3.1.1.15	≤ [1.92] psig

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

LCO 3.4.8 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

-----NOTES-----

1. Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period.
  2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.
- 

APPLICABILITY: MODE 3, with reactor steam dome pressure < [the RHR cut in permissive pressure].

ACTIONS

-----NOTE-----

LCO 3.0.4 is not applicable.

-----

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Initiate action to restore RHR shutdown cooling subsystem(s) to OPERABLE status.	Immediately
	<u>AND</u>	(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

LCO 3.4.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

-----NOTES-----

1. Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period.
  2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.
- 

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR) - High Water Level

LCO 3.9.8 One RHR shutdown cooling subsystem shall be OPERABLE and in operation.

-----NOTE-----  
The required RHR shutdown cooling subsystem may be removed from operation for up to 2 hours per 8 hour period.  
-----

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel and with the water level  $\geq$  [23] ft above the top of the [reactor pressure vessel (RPV) flange].

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required RHR shutdown cooling subsystem inoperable.	A.1 Verify an alternate method of decay heat removal is available.	1 hour <u>AND</u> Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Suspend loading irradiated fuel assemblies into the RPV.  <u>AND</u>	Immediately  (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Initiate action to restore [secondary] containment to OPERABLE status.	Immediately
	<u>AND</u>	
	B.3 Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
	<u>AND</u>	
C. Required RHR shutdown cooling subsystem not in operation.	C.1 Establish reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation
	<u>AND</u>	
	C.2 Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.8.1    Verify one RHR shutdown cooling subsystem is operating.	12 hours

3.9 REFUELING OPERATIONS

3.9.9 Residual Heat Removal (RHR) - Low Water Level

LCO 3.9.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and one RHR shutdown cooling subsystem shall be in operation.

-----NOTE-----  
The required operating shutdown cooling subsystem may be removed from operation for up to 2 hours per 8 hour period.  
-----

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel and with the water level < [23] ft above the top of the [reactor pressure vessel flange].

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to restore [secondary] containment to OPERABLE status.  <u>AND</u>	Immediately   (continued)



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
	<u>AND</u> B.3 Initiate action to restore one [secondary] containment isolation valve and associated instrumentation to OPERABLE status in each associated penetration flow path not isolated.	Immediately
C. No RHR shutdown cooling subsystem in operation.	C.1 Establish reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation
	<u>AND</u> C.2 Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.9.1 Verify one RHR shutdown cooling subsystem is operating.	12 hours

### 3.10 SPECIAL OPERATIONS

#### 3.10.8 SHUTDOWN MARGIN (SDM) Test - Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

- a. LCO 3.3.1.1, "Reactor Protection System Instrumentation," MODE 2 requirements for Function 2.a and 2.d of Table 3.3.1.1-1;
- b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with the banked position withdrawal sequence requirements of SR 3.3.2.1.8 changed to require the control rod sequence to conform to the SDM test sequence,

OR

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;
- c. Each withdrawn control rod shall be coupled to the associated CRD;
- d. All control rod withdrawals [during out of sequence control rod moves] shall be made in notch out mode; and
- e. No other CORE ALTERATIONS are in progress.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the above requirements not met, for reasons other than Condition B.	A.1 Place the reactor mode switch in the shutdown or refuel position.	Immediately
B. One control rod not coupled to its associated CRD.	B.1 Declare the affected control rod inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.10.8.1 Perform the applicable SRs for LCO 3.3.1.1, Functions 2.a and 2.d.	According to the applicable SRs
SR 3.10.8.2 -----NOTE----- Not required to be met if SR 3.10.8.3 satisfied. ----- Perform the applicable SRs for LCO 3.3.2.1, Function 2.	According to the applicable SRs

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.10.8.3 -----NOTE----- Not required to be met if SR 3.10.8.2 satisfied. -----</p> <p>Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff.</p>	<p>During control rod movement</p>
<p>SR 3.10.8.4 Verify no other CORE ALTERATIONS are in progress.</p>	<p>12 hours</p>
<p>SR 3.10.8.5 Verify each withdrawn control rod does not go to the overtravel position.</p>	<p>Each time the control rod is withdrawn to "full out" position</p> <p><u>AND</u></p> <p>Prior to satisfying LCO 3.10.8.c requirement after work on control rod or CRD System that could affect coupling</p>

5.0 ADMINISTRATIVE CONTROLS

5.6 Technical Specifications (TS) Bases Control

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- 5.6.1 Changes to the Bases of the TS shall be made under appropriate administrative controls and reviewed according to Specification 5.5.1.
- 5.6.2 Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
- a. A change in the TS incorporated in the license; or
  - b. A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- 5.6.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- 5.6.4 Proposed changes that meet the criteria of Specification 5.6.2.a or Specification 5.6.2.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71.
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5.0 ADMINISTRATIVE CONTROLS

5.7 Procedures, Programs, and Manuals

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5.7.1 Procedures

5.7.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
- b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in [Generic Letter 82-33];
- c. Security plan implementation;
- d. Emergency plan implementation;
- e. Quality assurance for effluent and environmental monitoring;
- f. Fire Protection Program implementation; and
- g. All programs specified in Specification 5.7.1.1.

5.7.1.2 Review and Approval

Each procedure of Specification 5.7.1.1, and changes thereto, shall be reviewed in accordance with Specification 5.5.1, approved by the [Plant Superintendent] or his designee in accordance with approved administrative procedures prior to implementation, except as specified in Specification 5.7.1.3, and reviewed periodically as set forth in administrative procedures.

5.7.1.3 Temporary Changes

Temporary changes to procedures of Specification 5.7.1.1 may be made provided:

- a. The intent of the existing procedure is not altered;
- b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator license on the unit affected; and

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.2 Process Control Program (FCP) (continued)

- b. Shall be effective after review and acceptance by the [review method of Specification 5.5.1] and the approval of the [Plant Superintendent].

5.7.2.3 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program; and
- b. The ODCM shall also contain the Radioactive Effluent Controls and Radiological Environmental Monitoring programs required by Specification 5.7.2.7 and Specification 5.7.2.8, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification [5.9.1.3] and Specification [5.9.1.4].

Licensee initiated changes to the ODCM:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
  - 1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and
  - 2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
- b. Shall become effective after review and acceptance by the [review method of Specification 5.5.1] and the approval of the [Plant Superintendent]; and

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.13 Ventilation Filter Testing Program (VFTP) (continued)

- a. Demonstrate for each of the ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [ $\pm 10\%$ ].

ESF Ventilation System	Flowrate
[ ]	[ ]

- b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [ $\pm 10\%$ ].

ESF Ventilation System	Flowrate
[ ]	[ ]

- c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of  $\leq$  [30°C] and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
[ ]	[ ]	[ ]

(continued)



5.7 Procedures, Programs, and Manuals

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5.7.2.15 Diesel Fuel Oil Testing Program (continued)

acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
  1. an API gravity or an absolute specific gravity within limits,
  2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
  3. a clear and bright appearance with proper color;
- b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and
- c. Total particulate concentration of the fuel oil is  $\leq 10$  mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.7.2.16 Fire Protection Program

This program provides controls to ensure that appropriate fire protection measures are maintained to protect the plant from fire and to ensure the capability to achieve and maintain safe shutdown in the event of a fire is maintained.

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BASES

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SAFETY LIMIT  
VIOLATIONS  
(continued)

2.2.3

If any SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 6). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations and the [offsite reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.5

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

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
  2. NEDE-24011-P-A (latest approved revision).
  3. XN-NF524(A), Revision 1, November 1983.
  4. 10 CFR 50.72.
  5. 10 CFR 100.
  6. 10 CFR 50.73.
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## B 2.0 SAFETY LIMITS (SLs)

### B 2.1.2 Reactor Coolant System (RCS) Pressure SL

#### BASES

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#### BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core is done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

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

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(continued)

BASES (continued)

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SAFETY LIMIT  
VIOLATIONS

2.2.1

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

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action.

2.2.3

If any SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations, and the [offsite reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.5

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

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(continued)

BASES

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LCO 3.0.2  
(continued)

ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

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

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.8.1, "AC Sources - Operating."

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

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

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(continued)

BASES

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SR 3.0.3  
(continued)

period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the

(continued)

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BASES

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SR 3.0.3  
(continued)

Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. This Specification applies to changes in MODES or other specified conditions in the Applicability associated with unit shutdown as well as startup.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

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BASES

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ACTIONS

A.1 (continued)

may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

If MFLPD cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared to F RTP or APRM gain or setpoints to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than F RTP, the appropriate gain or setpoint, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER  $\geq$  25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

(continued)



BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification than if MFLPD is less than or equal to fraction of rated power (FRP). When MFLPD is greater than FRP, more rapid changes in power distribution are typically expected.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23.
  2. FSAR, Section [ ].
  3. FSAR, Section [ ].
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## BASES

APPLICABLE  
SAFETY ANALYSES,  
LCO, and  
APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux-High,  
Setdown (continued)

low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux-High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux-High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux-High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux-High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux-High, Setdown Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux-High, Setdown with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux-High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

(continued)

## BASES

APPLICABLE  
SAFETY ANALYSES,  
LCO, and  
APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux-High,  
Setdown (continued)

In MODE 1, the Average Power Range Monitor Neutron Flux-High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated  
Thermal Power-High

The Average Power Range Monitor Flow Biased Simulated Thermal Power-High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor Fixed Neutron Flux-High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power-High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Fixed Neutron Flux-High Function will provide a scram signal before the Average Power Range Monitor Flow Biased Simulated Thermal Power-High Function setpoint is exceeded.

The APRM System is divided into two groups of channels with four APRM inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of

(continued)

BASES

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LCO  
(continued)

common discharge piping. Thus, to meet the LCO, both pumps in one loop or one pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

Note 1 permits both RHR shutdown cooling subsystems to be shut down for a period of 2 hours in an 8 hour period. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

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APPLICABILITY

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure above [the RHR cut in permissive pressure], this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures above the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS - Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

(continued)

BASES

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LCO  
(continued)

discharge piping. Thus, to meet the LCO, both pumps in one loop or one pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. In MODE 4, the RHR cross tie valve (2E11-F010) may be opened to allow pumps in one loop to discharge through the opposite recirculation loop to make a complete subsystem. Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

Note 1 permits both RHR shutdown cooling subsystems to be shut down for a period of 2 hours in an 8 hour period. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

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APPLICABILITY

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure above the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures above the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS - Operating") do not allow

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B 3.9 REFUELING OPERATIONS

B 3.9.8 Residual Heat Removal (RHR) - High Water Level

BASES

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BACKGROUND

The purpose of the RHR System in MODE 5 is to remove decay heat and sensible heat from the reactor coolant, as required by GDC 34. Each of the two shutdown cooling loops of the RHR System can provide the required decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after it has been cooled by circulation through the respective heat exchangers, to the reactor via the associated recirculation loop or to the reactor via the low pressure coolant injection path. The RHR heat exchangers transfer heat to the RHR Service Water System (LCO 3.7.1). The RHR shutdown cooling mode is manually controlled.

In addition to the RHR subsystems, the volume of water above the reactor pressure vessel (RPV) flange provides a heat sink for decay heat removal.

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APPLICABLE  
SAFETY ANALYSES

With the unavailability of MODE 5, the RHR System is not required to mitigate any events or accidents evaluated in the safety analyses. The RHR System is required for removing decay heat to maintain the temperature of the reactor coolant.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

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LCO

Only one RHR shutdown cooling subsystem is required to be OPERABLE and in operation in MODE 5 with the water level  $\geq$  [23] ft above the RPV flange. Only one subsystem is required because the volume of water above the RPV flange provides backup decay heat removal capability.

(continued)

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BASES

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LCO  
(continued)

An OPERABLE RHR shutdown cooling subsystem consists of an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. In MODE 5, the RHR cross tie valve is not required to be closed; thus, the valve may be opened to allow pumps in one loop to discharge through the opposite loop's heat exchanger to make a complete subsystem.

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. A Note is provided to allow a 2 hour exception to shut down the operating subsystem every 8 hours.

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APPLICABILITY

One RHR shutdown cooling subsystem must be OPERABLE and in operation in MODE 5, with irradiated fuel in the reactor pressure vessel and with the water level  $\geq$  [23] feet above the top of the RPV flange, to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR System requirements in MODE 5 with irradiated fuel in the reactor pressure vessel and with the water level  $<$  [23] ft above the RPV flange are given in LCO 3.9.9.

---

ACTIONS

A.1

With no RHR shutdown cooling subsystem OPERABLE, an alternate method of decay heat removal must be established within 1 hour. In this condition, the volume of water above the RPV flange provides adequate capability to remove decay heat from the reactor core. However, the overall reliability is reduced because loss of water level could

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BASES

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ACTIONS

A.1 (continued)

result in reduced decay heat removal capability. The 1 hour Completion Time is based on decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will ensure continued heat removal capability.

Alternate decay heat removal methods are available to the operators for review and preplanning in the unit's Operating Procedures. For example, this may include the use of the Reactor Water Cleanup System, operating with the regenerative heat exchanger bypassed. The method used to remove the decay heat should be the most prudent choice based on unit conditions.

B.1, B.2, B.3, and B.4

If no RHR subsystem is OPERABLE and an alternate method of decay heat removal is not available in accordance with Required Action A.1, actions shall be taken immediately to suspend operations involving an increase in reactor decay heat load by suspending loading of irradiated fuel assemblies into the RPV.

Additional actions are required to minimize any potential fission product release to the environment. This includes initiating immediate action to restore the following to OPERABLE status: secondary containment, one standby gas treatment subsystem, and one secondary containment isolation valve and associated instrumentation in each associated penetration not isolated. This may be performed as an administrative check, by examining logs or other information to determine whether the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, a surveillance may need to be performed to restore the component to OPERABLE status.

(continued)



BASES

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ACTIONS

B.1, B.2, B.3, and B.4 (continued)

Actions must continue until all required components are OPERABLE.

C.1 and C.2

If no RHR Shutdown Cooling System is in operation, an alternate method of coolant circulation is required to be established within 1 hour. The Completion Time is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation.

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR Shutdown Cooling System), the reactor coolant temperature must be periodically monitored to ensure proper functioning of the alternate method. The once per hour Completion Time is deemed appropriate.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.8.1

This Surveillance demonstrates that the RHR subsystem is in operation and circulating reactor coolant.

The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

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REFERENCES

None.

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BASES

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LCO  
(continued)

allow pumps in one loop to discharge through the opposite loop's heat exchanger to make a complete subsystem.

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. A Note is provided to allow a 2 hour exception to shut down the operating subsystem every 8 hours.

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APPLICABILITY

Two RHR shutdown cooling subsystems are required to be OPERABLE, and one must be in operation in MODE 5, with irradiated fuel in the reactor pressure vessel and with the water level  $< [23]$  ft above the top of the RPV flange, to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR System requirements in MODE 5 with irradiated fuel in the reactor pressure vessel and with the water level  $\geq [23]$  ft above the RPV flange are given in LCO 3.9.8, "Residual Heat Removal (RHR) - High Water Level."

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ACTIONS

A.1

With one of the two required RHR shutdown cooling subsystems inoperable, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore an alternate method of decay heat removal must be provided. With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the

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BASES

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ACTIONS

A.1 (continued)

available decay heat removal capabilities. Furthermore, verification of the functional availability of this alternate method(s) must be reconfirmed every 24 hours thereafter. This will ensure continued heat removal capability.

Alternate decay heat removal methods are available to the operators for review and preplanning in the unit's Operating Procedures. For example, this may include the use of the Reactor Water Cleanup System, operating with the regenerative heat exchanger bypassed. The method used to remove decay heat should be the most prudent choice based on unit conditions.

B.1, B.2, and B.3

With the required decay heat removal subsystem(s) inoperable and the required alternate method(s) of decay heat removal not available in accordance with Required Action A.1, additional actions are required to minimize any potential fission product release to the environment. This includes initiating immediate action to restore the following to OPERABLE status: secondary containment, one standby gas treatment subsystem, and one secondary containment isolation valve and associated instrumentation in each associated penetration not isolated. This may be performed as an administrative check, by examining logs or other information to determine whether the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the surveillance may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

C.1 and C.2

If no RHR subsystem is in operation, an alternate method of coolant circulation is required to be established within 1 hour. The Completion Time is modified such that the

(continued)

BASES

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ACTION

C.1 and C.2 (continued)

1 hour is applicable separately for each occurrence involving a loss of coolant circulation.

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR Shutdown Cooling System), the reactor coolant temperature must be periodically monitored to ensure proper functioning of the alternate method. The once per hour Completion Time is deemed appropriate.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.9.1

This Surveillance demonstrates that one RHR subsystem is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability.

The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystems in the control room.

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REFERENCES

None.

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1 and 2 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1 and 2 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, are required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1 and 2). In addition to the added requirements for the RWM, IRM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

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LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. Because multiple control rods will be withdrawn and the reactor will potentially become critical, RPS MODE 2 requirements for Functions 2.a and 2.d of Table 3.3.1.1-1 must be enforced and the approved control rod withdrawal sequence must be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2), or must be verified by a second licensed operator or other

(continued)

BASES

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LCO  
(continued)

qualified member of the technical staff. To provide additional protection against an inadvertent criticality, control rod withdrawals that do not conform to the banked position withdrawal sequence specified in LCO 3.1.6, "Rod Pattern Control," (i.e., out of sequence control rod withdrawals) must be made in the individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Coupling integrity of withdrawn control rods is required to minimize the probability of a CRDA and ensure proper functioning of the withdrawn control rods, if they are required to scram. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATIONS may be in progress. This Special Operations LCO then allows changing the Table 1.1-1 reactor mode switch position requirements to include the startup/hot standby position, such that the SDM tests may be performed while in MODE 5.

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APPLICABILITY

These SDM test Special Operations requirements are only applicable if the SDM tests are to be performed while in MODE 5 with the reactor vessel head removed or the head bolts not fully tensioned. Additional requirements during these tests to enforce control rod withdrawal sequences and restrict other CORE ALTERATIONS provide protection against potential reactivity excursions. Operations in all other MODES are unaffected by this LCO.

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ACTIONS

A.1

With one or more of the requirements of this LCO not met, the testing should be immediately stopped by placing the reactor mode switch in the shutdown or refuel position. This results in a condition that is consistent with the requirements for MODE 5 where the provisions of this Special Operations LCO are no longer required.

B.1

With the requirements of this LCO not met, the affected control rod shall be declared inoperable. This results in a condition that is consistent with the requirements for MODE

(continued)

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BASES

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ACTIONS

B.1 (continued)

5 where the provisions of this Special Operations LCO are no longer required.

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SURVEILLANCE  
REQUIREMENTS

SR 3.10.8.1

Performance of the applicable SRs for LCO 3.3.1.1, Functions 2.a and 2.d will ensure that the reactor is operated within the bounds of the safety analysis.

SR 3.10.8.2 and SR 3.10.8.3

The control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff. As noted, either the applicable SRs for the RWM (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.1), or the proper movement of control rods must be verified (SR 3.10.8.2). This latter verification (i.e., SR 3.10.8.2) must be performed during control rod movement to prevent deviations from the specified sequence. These surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.10.8.5 (continued)

rod is withdrawn to the "full out" notch position, or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

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REFERENCES

1. NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel, Supplement for United States (as amended).
  2. Letter from T. Pickens (BWROG) to G.C. Lainas, NRC, "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," August 15, 1986.
  3. [Plant specific transient analysis].
  4. [Plant specific reload analysis].
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# **BWR/6**

**NUREG 1434**

## **AFFECTED BY CHANGE PACKAGES**

**BWR-01  
BWR-01A  
BWR-02  
BWR-03  
WOG-01  
WOG-06  
BWOG-01**

## STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWR-01  
NUREG 1434

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

---

Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

---

Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWR-01A  
NUREG 1434

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Technical Specification  
Subcommittee Chairman

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWR-02  
NUREG 1434

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Technical Specification  
Subcommittee Chairman

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Technical Specification  
Subcommittee Chairman

## STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWR-03  
NUREG 1434

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-01  
NUREG 1434

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Technical Specification  
Subcommittee Chairman

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Technical Specification  
Subcommittee Chairman

## STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-06  
NUREG 1434

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

## STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWOG-01  
NUREG 1434

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman



1.0 USE AND APPLICATION

1.1 Definitions

-----NOTE-----

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

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
AVERAGE PLANAR EXPOSURE	The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.
AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	The APLHGR shall be applicable to a specific planar height and is equal to the sum of the [LHGRs] [heat generation rate per unit length of fuel rod] for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle [at the height].
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place cross calibration of the sensing elements and normal calibration of the remaining adjustable devices in the channel. Whenever a sensing element is replaced, the next required in-place cross calibration consists of comparing the other sensing elements with the recently installed sensing element. The CHANNEL

(continued)

1.1 Definitions

---

CHANNEL CALIBRATION  
(continued)

CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

- a. Analog channels—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips.
- b. Bistable channels (e.g., pressure switches and switch contacts)—the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm and trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity, within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement) is not considered a CORE ALTERATION. In addition, control rod movement with other than the normal control rod drive is not considered a

(continued)

1.1 Definitions

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CORE ALTERATION  
(continued)

CORE ALTERATION provided there are no fuel assemblies in associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS  
REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.9.1.6. Plant operation within these limits is addressed in individual Specifications.

DOSE EQUIVALENT I-131

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

$\bar{E}$  - AVERAGE  
DISINTEGRATION ENERGY

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

EMERGENCY CORE COOLING  
SYSTEM (ECCS) RESPONSE  
TIME

The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading

(continued)

## 1.1 Definitions

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EMERGENCY CORE COOLING  
SYSTEM (ECCS) RESPONSE  
TIME  
(continued)

delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

END OF CYCLE  
RECIRCULATION PUMP TRIP  
(EOC-RPT) SYSTEM RESPONSE  
TIME

The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by [the associated turbine stop valve limit switch or from when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint] to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured, [except for the breaker arc suppression time, which is not measured but is validated to conform to the manufacturer's design value].

ISOLATION SYSTEM  
RESPONSE TIME

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or
2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE;

(continued)

1.1 Definitions

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LEAKAGE (continued)	<p>b. <u>Unidentified LEAKAGE</u></p> <p>All LEAKAGE into the drywell that is not identified LEAKAGE;</p> <p>c. <u>Total LEAKAGE</u></p> <p>Sum of the identified and unidentified LEAKAGE;</p> <p>d. <u>Pressure Boundary LEAKAGE</u></p> <p>LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.</p>
LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)	The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core [for each class of fuel]. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

(continued)

1.1 Definitions (continued)

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MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: <ul style="list-style-type: none"> <li>a. Described in Chapter [14, Initial Test Program] of the FSAR;</li> <li>b. Authorized under the provisions of 10 CFR 50.59; or</li> <li>c. Otherwise approved by the Nuclear Regulatory Commission.</li> </ul>
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.9.1.7. Plant operation within these operating limits is addressed in LCO 3.4.11, "RCS Pressure and Temperature (P/T) Limits."
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of [3833] MWt.

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(continued)

1.1 Definitions (continued)

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REACTOR PROTECTION  
SYSTEM (RPS) RESPONSE  
TIME

The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

- a. The reactor is xenon free;
- b. The moderator temperature is 68°F; and
- c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn.

With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during  $n$  Surveillance Frequency intervals, where  $n$  is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER

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

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(continued)

## 1.1 Definitions (continued)

TURBINE BYPASS SYSTEM  
RESPONSE TIME

The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:

- a. The time for initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and
- b. The time for initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.



## 1.0 USE AND APPLICATION

### 1.2 Logical Connectors

---

**PURPOSE** The purpose of this section is to explain the meaning of logical connectors.

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

---

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

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

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**EXAMPLES** The following examples illustrate the use of logical connectors.

(continued)

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## 1.0 USE AND APPLICATION

### 1.3 Completion Times

---

---

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

---

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

---

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

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

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

(continued)

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1.3 Completion Times

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DESCRIPTION  
(continued)

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

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

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

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

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

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

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(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)

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1.3 Completion Times

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## EXAMPLES

EXAMPLE 1.3-3 (continued)

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

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

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

(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	<u>AND</u> B.2 Be in MODE 4.	36 hours

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

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

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

(continued)

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

---

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

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

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

---

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

- a. MODE 2 within 7 hours;
- b. MODE 3 within 13 hours; and
- c. MODE 4 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

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

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

---

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This

(continued)

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### 3.0 LCO APPLICABILITY

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LCO 3.0.4  
(continued)      Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

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

---

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

---

LCO 3.0.6      When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, additional evaluations and limitations may be required in accordance with Specification 5.8, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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

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(continued)



3.0 LCO APPLICABILITY (continued)

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LCO 3.0.7 Special Operations LCOs in Section 3.10 allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain LCO 3.0.7 unchanged. Compliance with Special Operations LCOs is optional. When a Special Operations LCO is desired to be met but is not met, the ACTIONS of the Special Operations LCO shall be met. When a Special Operations LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

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### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

---

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

---

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

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

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

Exceptions to this Specification are stated in the individual Specifications.

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

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

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be

(continued)

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3.0 SR APPLICABILITY

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SR 3.0.3            declared not met, and the applicable Condition(s) must be  
(continued)        entered.

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Sk 3.0.4            Entry into a MODE or other specified condition in the  
                      Applicability of an LCO shall not be made unless the LCO's  
                      Surveillances have been met within their specified  
                      Frequency. This provision shall not prevent passage through  
                      or to MODES or other specified conditions in compliance with  
                      Required Actions.

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.2.4.1 -----NOTE-----                      Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 Item b or c requirements.                      -----                      Verify MFLPD is within limits.</p>	<p>Once within 12 hours after <math>\geq 25\%</math> RTP   <u>AND</u>                      24 hours thereafter</p>
<p>SR 3.2.4.2 -----NOTE-----                      Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 Item a requirements.                      -----                      Verify APRM setpoints or gains are adjusted for the calculated MFLPD.</p>	<p>12 hours</p>

Table 3.3.1.1-1 (page 1 of 3)  
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	[3]	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ [122/125] divisions of full scale
	5(a)	[3]	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ [122/125] divisions of full scale
b. Inop	2	[3]	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	[3]	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	[3]	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ [20]% RTP
b. Flow Biased Simulated Thermal Power - High	1	[3]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ [0.66 W + 67]% RTP and ≤ [113]% RTP[(b)]

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Allowable Value is (≤ 0.66 W + 43%) RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

LCO 3.4.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

-----NOTES-----

1. Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period.
  2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.
- 

APPLICABILITY: MODE 3 with reactor steam dome pressure < [the RHR cut in permissive pressure].

ACTIONS

-----NOTE-----

LCO 3.0.4 is not applicable.

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CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Initiate action to restore RHR shutdown cooling subsystem(s) to OPERABLE status.	Immediately
	<u>AND</u>	(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

LCO 3.4.10 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

-----NOTES-----

1. Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period.
2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

-----

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR) - High Water Level

LCO 3.9.8 One RHR shutdown cooling subsystem shall be OPERABLE and in operation.

-----NOTE-----  
The required RHR shutdown cooling subsystem may be removed from operation for up to 2 hours per 8 hour period.  
-----

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel and with the water level  $\geq$  [22 ft 8 inches] above the top of the [reactor pressure vessel (RPV) flange].

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required RHR shutdown cooling subsystem inoperable.	A.1 Verify an alternate method of decay heat removal is available.	1 hour <u>AND</u> Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Suspend loading irradiated fuel assemblies into the RPV.  <u>AND</u>	Immediately    (continued)



ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Initiate action to restore [primary or secondary] containment to OPERABLE status.	Immediately
	<u>AND</u>	
	B.3 Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
<u>AND</u>		
	B.4 Initiate action to restore one secondary containment isolation valve and associated instrumentation to OPERABLE status in each associated penetration flow path not isolated.	Immediately
C. Required RHR shutdown cooling subsystem not in operation.	C.1 Establish reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation
	<u>AND</u>	
	C.2 Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.8.1    Verify one RHR shutdown cooling subsystem is operating.	12 hours

3.9 REFUELING OPERATIONS

3.9.9 Residual Heat Removal (RHR) - Low Water Level

LCO 3.9.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and one RHR shutdown cooling subsystem shall be in operation.

-----NOTE-----  
The required operating shutdown cooling subsystem may be removed from operation for up to 2 hours per 8 hour period.  
-----

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel and with the water level < [23] ft above the top of the [reactor pressure vessel flange].

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to restore [primary or secondary] containment to OPERABLE status. <u>AND</u>	Immediately  (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
	<u>AND</u> B.3 Initiate action to restore one secondary containment isolation valve and associated instrumentation to OPERABLE status in each associated penetration flow path not isolated.	Immediately
C. No RHR shutdown cooling subsystem in operation.	C.1 Establish reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation
	<u>AND</u> C.2 Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.9.1 Verify one RHR shutdown cooling subsystem is operating.	12 hours

5.0 ADMINISTRATIVE CONTROLS

5.6 Technical Specifications (TS) Bases Control

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- 5.6.1 Changes to the Bases of the TS shall be made under appropriate administrative controls and reviewed according to Specification 5.5.1.
- 5.6.2 Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
- a. A change in the TS incorporated in the license; or
  - b. A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- 5.6.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- 5.6.4 Proposed changes that meet the criteria of Specification 5.6.2.a or Specification 5.6.2.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71.
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5.0 ADMINISTRATIVE CONTROLS

5.7 Procedures, Programs, and Manuals

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5.7.1 Procedures

5.7.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
- b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in [Generic Letter 82-33];
- c. Security plan implementation;
- d. Emergency plan implementation;
- e. Quality assurance for effluent and environmental monitoring;
- f. Fire Protection Program implementation; and
- g. All programs specified in Specification 5.7.2.

5.7.1.2 Review and Approval

Each procedure of Specification 5.7.1.1, and changes thereto, shall be reviewed in accordance with Specification 5.5.1, approved by the [Plant Superintendent] or his designee in accordance with approved administrative procedures prior to implementation, except as specified in Specification 5.7.1.3, and reviewed periodically as set forth in administrative procedures.

5.7.1.3 Temporary Changes

Temporary changes to procedures of Specification 5.7.1.1 may be made provided:

- a. The intent of the existing procedure is not altered;
- b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator license on the unit affected; and

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.2 Process Control Program (PCP) (continued)

- b. Shall be effective after review and acceptance by the [review method of Specification 5.5.1] and the approval of the [Plant Superintendent].

5.7.2.3 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program; and
- b. The ODCM shall also contain the Radioactive Effluent Controls and Radiological Environmental Monitoring programs required by Specification 5.7.2.7 and Specification 5.7.2.8, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification [5.9.1.3] and Specification [5.9.1.4].

Licensee initiated changes to the ODCM:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
  - 1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and
  - 2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
- b. Shall become effective after review and acceptance by the [review method of Specification 5.5.1] and the approval of the [Plant Superintendent]; and

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.13 Ventilation Filter Testing Program (VFTP) (continued)

- a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass  $< [0.05]\%$  when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below  $[\pm 10\%]$ :

ESF Ventilation System	Flowrate
[ ]	[ ]

- b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass  $< [0.05]\%$  when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below  $[\pm 10\%]$ :

ESF Ventilation System	Flowrate
[ ]	[ ]

- c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in [Regulatory Guide 1.52, Revision 2], shows the methyl iodide penetration less than the value specified below when tested in accordance with [ASTM D3803-1989] at a temperature of  $\leq [30^{\circ}\text{C}]$  and greater than or equal to the relative humidity specified below:

ESF Ventilation System	Penetration	RH
[ ]	[ ]	[ ]

(continued)



5.7 Procedures, Programs, and Manuals

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5.7.2 Programs and Manuals (continued)

5.7.2.15 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
  1. an API gravity or an absolute specific gravity within limits,
  2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil,
  3. a clear and bright appearance with proper color;
- b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and
- c. Total particulate concentration of the fuel oil is  $\leq 10$  mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.7.2.16 Fire Protection Program

This program provides controls to ensure that appropriate fire protection measures are maintained to protect the plant from fire and to ensure the capability to achieve and maintain safe shutdown in the event of a fire is maintained.

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BASES

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SAFETY LIMIT  
VIOLATIONS  
(continued)

2.2.3

If any SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 [Ref. 6]. A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations and the [offsite reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.5

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

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
  2. NEDE-24011-P-A, (latest approved revision).
  3. XN-NF524(A), Revision 1, November 1983.
  4. 10 CFR 50.72.
  5. 10 CFR 100.
  6. 10 CFR 50.73.
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B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

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BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core is done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

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

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(continued)

BASES

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SAFETY LIMIT  
VIOLATIONS  
(continued)

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action.

2.2.3

If any SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President - Nuclear Operations, and the [offsite reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.5

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

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

(continued)

BASES

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LCO 3.0.2  
(continued)

ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

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

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.8.1, "AC Sources - Operating."

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

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

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(continued)

BASES

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LCO 3.0.5  
(continued)

isolation valve that has been closed to comply with Required Actions, and must be reopened to perform the SRs.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

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LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the plant is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the support system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be

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BASES

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SR 3.0.3  
(continued)

period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable then is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion

(continued)

BASES

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SR 3.0.3  
(continued)

Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. This Specification applies to changes in MODES or other specified conditions in the Applicability associated with unit shutdown as well as startup.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

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BASES

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ACTIONS

A.1 (continued)

may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

B.1

If the APRM gain or setpoints cannot be restored to within their required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared to F RTP or APRM gain or setpoints to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are required only to determine the MFLPD and, assuming MFLPD is greater than F RTP, the appropriate gain or setpoint, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER  $\geq$  25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification than if MFLPD is less than or equal to fraction of rated power (FRP). When MFLPD is greater than FRP, more rapid changes in power distribution are typically expected.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 29.
  2. FSAR, Section [ ].
  3. FSAR, Section [ ].
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BASES

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LCO  
(continued)

associated piping and valves. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

Note 1 permits both RHR shutdown cooling subsystems and recirculation pumps to be shut down for a period of 2 hours in an 8 hour period. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR system in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

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APPLICABILITY

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure above the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures above the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the low pressure RHR shutdown cooling subsystem into operation.

In MODE 3 with reactor steam dome pressure below the RHR cut in permissive pressure (i.e., the actual pressure at which the interlock resets) the RHR System may be operated in the

(continued)

BASES

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LCO  
(continued)

pipng and valves. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

Note 1 permits both RHR shutdown cooling subsystems and recirculation pumps to be shut down for a period of 2 hours in an 8 hour period. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

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APPLICABILITY

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure above the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures above the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the low pressure RHR shutdown cooling subsystem into operation.

In MODE 4, the RHR System may be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 200°F.

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BASES

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LCO  
(continued)

An OPERABLE RHR shutdown cooling subsystem consists of an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path.

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. A Note is provided to allow a 2 hour exception to shut down the operating subsystem every 8 hours.

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APPLICABILITY

One RHR shutdown cooling subsystem must be OPERABLE in MODE 5, with irradiated fuel in the reactor pressure vessel and with the water level  $\geq$  [22 ft 8 inches] above the top of the RPV flange, to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR System requirements in MODE 5, with irradiated fuel in the reactor pressure vessel and with the water level  $<$  [22 ft 8 inches] above the RPV flange, are given in LCO 3.9.9, "Residual Heat Removal (RHR) - Low Water Level."

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ACTIONS

A.1

With no RHR shutdown cooling subsystem OPERABLE, an alternate method of decay heat removal must be established within 1 hour. In this condition, the volume of water above the RPV flange provides adequate capability to remove decay heat from the reactor core. However, the overall reliability is reduced because loss of water level could result in reduced decay heat removal capability. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be

(continued)

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BASES

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ACTIONS

A.1 (continued)

reconfirmed every 24 hours thereafter. This will ensure continued heat removal capability.

Alternate decay heat removal methods are available to the operators for review and preplanning in the unit's Operating Procedures. For example, this may include the use of the Reactor Water Cleanup System, operating with the regenerative heat exchanger bypassed. The method used to remove the decay heat should be the most prudent choice based on unit conditions.

B.1, B.2, B.3, and B.4

If no RHR shutdown cooling subsystem is OPERABLE and an alternate method of decay heat removal is not available in accordance with Required Action A.1, actions shall be taken immediately to suspend operations involving an increase in reactor decay heat load by suspending the loading of irradiated fuel assemblies into the RPV.

Additional actions are required to minimize any potential fission product release to the environment. This includes initiating immediate action to restore the following to OPERABLE status: secondary containment, one standby gas treatment subsystem, and one secondary containment isolation valve and associated instrumentation in each associated penetration not isolated. This may be performed as an administrative check, by examining logs or other information to determine whether the components are out of service for maintenance or other reasons. It does not mean to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, a surveillance may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

C.1 and C.2

If no RHR Shutdown Cooling System is in operation, an alternate method of coolant circulation is required to be established within 1 hour. The Completion Time is modified

(continued)

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BASES

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ACTIONS

C.1 and C.2 (continued)

such that 1 hour is applicable separately for each occurrence involving a loss of coolant circulation.

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR Shutdown Cooling System), the reactor coolant temperature must be periodically monitored to ensure proper functioning of the alternate method. The once per hour Completion Time is deemed appropriate.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.8.1

This Surveillance demonstrates that the RHR subsystem is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

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REFERENCES

None.

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B 3.9 REFUELING OPERATIONS

B 3.9.9 Residual Heat Removal (RHR) - Low Water Level

BASES

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BACKGROUND

The purpose of the RHR System in MODE 5 is to remove decay heat and sensible heat from the reactor coolant, as required by GDC 34. Each of the two shutdown cooling loops of the RHR System can provide the required decay heat removal. Each loop consists of one motor driven pump, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after it has been cooled by circulation through the respective heat exchangers, to the reactor via separate feedwater lines, to the upper containment pool via a common single flow distribution sparger, or to the reactor via the low pressure coolant injection path. The RHR heat exchangers transfer heat to the Standby Service Water System (LCO 3.7.1). The RHR shutdown cooling mode is manually controlled.

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APPLICABLE  
SAFETY ANALYSES

With the unit in MODE 5, the RHR System is not required to mitigate any events or accidents evaluated in the safety analyses. The RHR System is required for removing decay heat to maintain the temperature of the reactor coolant.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

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LCO

In MODE 5 with the water level < 22 ft 8 inches above the reactor pressure vessel (RPV) flange both RHR shutdown cooling subsystems must be OPERABLE.

An OPERABLE RHR shutdown cooling subsystem consists of an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path.

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or

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BASES

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LCO  
(continued) local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. A Note is provided to allow a 2 hour exception to shut down the operating subsystem every 8 hours.

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APPLICABILITY Two RHR shutdown cooling subsystems are required to be OPERABLE in MODE 5, with irradiated fuel in the reactor pressure vessel and with the water level < [22 ft 8 inches] above the top of the RPV flange, to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR System requirements in MODE 5, with irradiated fuel in the reactor pressure vessel and with the water level  $\geq$  [22 ft 8 inches] above the RPV flange, are given in LCO 3.9.8, "Residual Heat Removal (RHR) - High Water Level."

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ACTIONS

A.1

With one of the two required RHR shutdown cooling subsystems inoperable, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore an alternate method of decay heat removal must be provided. With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will ensure continued heat removal capability.

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BASES

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ACTIONS

A.1 (continued)

Alternate decay heat removal methods are available to the operators for review and preplanning in the unit's Operating Procedures. For example, this may include the use of the Reactor Water Cleanup System, operating with the regenerative heat exchanger bypassed. The method used to remove decay heat should be the most prudent choice based on unit conditions.

B.1, B.2, and B.3

With the required decay heat removal subsystem(s) inoperable and the required alternate method(s) of decay heat removal not available in accordance with Required Action A.1, additional actions are required to minimize any potential fission product release to the environment. This includes initiating immediate action to restore the following to OPERABLE status: secondary containment, one standby gas treatment subsystem, and one secondary containment isolation valve and associated instrumentation in each associated penetration not isolated. This may be performed as an administrative check, by examining logs or other information to determine whether the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the surveillance may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

C.1 and C.2

If no RHR shutdown cooling subsystem is in operation, an alternate method of coolant circulation is required to be established within 1 hour. The Completion Time is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation.

During the period when the reactor coolant is being circulated by an alternate method (other than by the

(continued)

BASES

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ACTIONS

C.1 and C.2 (continued)

required RHR Shutdown Cooling System), the reactor coolant temperature must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.9.1

This Surveillance demonstrates that one RHR subsystem is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

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REFERENCES

None.

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1 and 2 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1 and 2 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, are required to demonstrate that the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1 and 2). In addition to the added requirements for the RPC, IRM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

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LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. Because multiple control rods will be withdrawn and the reactor will potentially become critical, RPS MODE 2 requirements for Functions 2.a and 2.d of Table 3.3.1.1-1 must be enforced and the approved control rod withdrawal sequence must be enforced by the RPC (LCO 3.3.2.1, Function 1b, MODE 2), or must be verified by a second licensed operator or other

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BASES

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LCO  
(continued)

qualified member of the technical staff. To provide additional protection against an inadvertent criticality, control rod withdrawals that do not conform to the banked position withdrawal sequence specified in LCO 3.1.6, "Rod Pattern Control" (i.e., out of sequence control rod withdrawals) must be made in the notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Coupling integrity of withdrawn control rods is required to minimize the probability of a CRDA and ensure proper functioning of the withdrawn control rods, if they are required to scram. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATIONS may be in progress. This Special Operations LCO then allows changing the Table 1.1-1 reactor mode switch position requirements to include the startup/hot standby position, such that the SDM tests may be performed while in MODE 5.

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APPLICABILITY

These SDM test Special Operations requirements are only applicable if the SDM tests are to be performed while in MODE 5 with the reactor vessel head removed or the head bolts not fully tensioned. Additional requirements during these tests to enforce control rod withdrawal sequences and restrict other CORE ALTERATIONS provide protection against potential reactivity excursions. Operations in all other MODES are unaffected by this LCO.

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ACTIONS

A.1

With one or more of the requirements of this LCO not met, the testing should be immediately stopped by placing the reactor mode switch in the shutdown or refuel position. This results in a condition that is consistent with the requirements for MODE 5 where the provisions of this Special Operations LCO are no longer required.

B.1

With the requirements of this LCO not met, the affected control rod shall be declared inoperable. This results in a condition that is consistent with the requirements for MODE

(continued)

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BASES

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ACTIONS

B.1 (continued)

5 where the provisions of this Special Operations LCO are no longer required.

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SURVEILLANCE  
REQUIREMENTS

SR 3.10.8.1

Performance of the applicable SRs for LCO 3.3.1.1, Functions 2.a and 2.d will ensure that the reactor is operated within the bounds of the safety analysis.

SR 3.10.8.2 and SR 3.10.8.3

The control rod withdrawal sequences during the SDM tests may be enforced by the RPC (LCO 3.3.2.1, Function 1b, MODE 2 requirements) or by a second licensed operator or other qualified member of the technical staff. As noted, either the applicable SRs for the RPC (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.1 and SR 3.10.8.2), or the proper movement of control rods must be verified. This latter verification (i.e., SR 3.10.8.2) must be performed during control rod movement to prevent deviations from the specified sequence. These surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.10.8.5 (continued)

rod is withdrawn to the "full out" notch position or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

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REFERENCES

1. NEDE-24011-P-A-US, General Electric Standard Application for Reactor Fuel, Supplement For United States (as amended).
  2. Letter, T.A. Pickens (BWROG) to G.C. Lainas (NRC), "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," August 15, 1986.
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B 3.10 SPECIAL OPERATIONS

B 3.10.9 Recirculation Loops - Testing

BASES

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BACKGROUND

The purpose of this Special Operations LCO in MODES 1 and 2 is to allow either the PHYSICS TESTS or the Startup Test Program to be performed with less than two recirculation loops in operation. Testing performed as part of the Startup Test Program (Ref. 1), or PHYSICS TESTS authorized under the provisions of 10 CFR 50.59 (Ref. 2) or otherwise approved by the NRC, may be required to be performed under natural circulation conditions with the reactor critical. LCO 3.4.1, "Recirculation Loops Operating," requires that one or both recirculation loops be in operation during MODES 1 and 2. This Special Operations LCO provides the appropriate additional restrictions to allow testing at natural circulation conditions or in single loop operation with the reactor critical.

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APPLICABLE  
SAFETY ANALYSES

The operation of the Reactor Coolant Recirculation System is an initial condition assumed in the design basis loss of coolant accident (Ref. 3). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the postulated accident. During PHYSICS TESTS  $\leq$  5% RTP, or limited testing during the Startup Test Program for the initial cycle, the decay heat in the reactor coolant is sufficiently low, such that the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important. In addition, the probability of a Design Basis Accident (DBA) or other accidents occurring during the limited time allowed at natural circulation or in single loop operation is low.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

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(continued)



BASES (continued)

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LCO As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. However, to perform testing at natural circulation conditions or with a single operating loop, operations must be limited to those tests defined in the Startup Test Program or approved PHYSICS TESTS performed at  $\leq$  [5%] RTP. To minimize the probability of an accident, while operating at natural circulation conditions or with one operating loop, the duration of these tests is limited to  $\leq$  24 hours. This Special Operations LCO then allows suspension of the requirements of LCO 3.4.1 during such testing. In addition to the requirements of this LCO, the normally required MODE 1 or MODE 2 applicable LCOs must be met.

---

APPLICABILITY This Special Operations LCO may only be used while performing testing at natural circulation conditions or while operating with a single loop, as may be required as part of the Startup Test Program or during low power PHYSICS TESTS. Additional requirements during these tests to limit the operating time at natural circulation conditions reduce the probability that a DBA may occur with both recirculation loops not in operation. Operations in all other MODES are unaffected by this LCO.

---

ACTIONS A.1

With the testing performed at natural circulation conditions or with a single operating loop, and the duration of the test exceeding the 24 hour time limit, actions should be taken to promptly shut down. Inserting all insertable control rods will result in a condition that does not require both recirculation loops to be in operation. The allowed Completion Time of [1] hour provides sufficient time to insert the withdrawn control rods.

B.1

With the requirements of this LCO not met for reasons other than those specified in Condition A (e.g., low power PHYSICS TESTS exceeding [5%] RTP, or unapproved testing at natural circulation), the reactor mode switch should immediately be

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BASES

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ACTIONS

B.1 (continued)

placed in the shutdown position. This results in a condition that does not require both recirculation loops to be in operation. The action to immediately place the reactor mode switch in the shutdown position prevents unacceptable consequences from an accident initiated from outside the analysis bounds. Also, operation beyond authorized bounds should be terminated upon discovery.

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SURVEILLANCE  
REQUIREMENTS

SR 3.10.9.1 and SR 3.10.9.2

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of this LCO. Because the 1 hour Frequency provides frequent checks of the LCO requirements during the allowed 24 hour testing interval, the probability of operation outside the limits concurrent with a postulated accident is reduced even further.

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REFERENCES

1. FSAR, Chapter [14].
  2. 10 CFR 50.59.
  3. FSAR, Section [6.3.3.4].
  4. FSAR, Section [ ].
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B 3.10 SPECIAL OPERATIONS

B 3.10.10 Training Startups

BASES

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BACKGROUND

The purpose of this Special Operations LCO is to permit training startups to be performed while in MODE 2 to provide plant startup experience for reactor operators. This training involves withdrawal of control rods to achieve criticality and then further withdrawal of control rods, as would be experienced during an actual plant startup. During these training startups, if the reactor coolant is allowed to heat up, maintenance of a constant reactor vessel water level requires the passage of reactor coolant through the Reactor Water Cleanup System, as the reactor coolant specific volume increases. Since this results in reactor water discharge to the radioactive waste disposal system, the amount of this discharge should be minimized. This Special Operations LCO provides the appropriate additional controls to allow one residual heat removal (RHR) subsystem to be aligned in the shutdown cooling mode, so that the reactor coolant temperature can be controlled during the training startups, thereby minimizing the discharge of reactor water to the radioactive waste disposal system.

---

APPLICABLE  
SAFETY ANALYSES

The Emergency Core Cooling System (ECCS) is designed to provide core cooling following a loss of coolant accident (LOCA). The low pressure coolant injection (LPCI) mode of the RHR System is one of the ECCS subsystems assumed to function during a LOCA. With reactor power  $\leq 1\%$  RTP (equivalent to all OPERABLE intermediate range monitor (IRM) channels  $\leq 25/40$  divisions of full scale on Range 7) and average reactor coolant temperature  $< 200^\circ\text{F}$ , the stored energy in the reactor core and coolant system is very low, and a reduced complement of ECCS can provide the required core cooling, thereby allowing operation with one RHR subsystem in the shutdown cooling mode (Ref. 1).

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A

(continued)

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BASES

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APPLICABLE SAFETY ANALYSES (continued) discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

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LCO As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Training startups may be performed while in MODE 2 with no RHR subsystems aligned in the shutdown cooling mode and, therefore, without meeting this Special Operations LCO or its ACTIONS. However, to minimize the discharge of reactor coolant to the radioactive waste disposal system, performance of the training startups may be performed with one RHR subsystem aligned in the shutdown cooling mode to maintain reactor coolant temperature < 200°F. Under these conditions, the THERMAL POWER must be maintained  $\leq 1\%$  RTP (equivalent to all OPERABLE IRM channels  $\leq 25/40$  divisions of full scale on Range 7) and the reactor coolant temperature must be  $\leq 200^\circ\text{F}$ . This Special Operations LCO then allows changing the LPCI OPERABILITY requirements. In addition to the requirements of this LCO, the normally required MODE 2 applicable LCOs must also be met.

---

APPLICABILITY Training startups while in MODE 2 may be performed with one RHR subsystem aligned in the shutdown cooling mode to control the reactor coolant temperature. Additional requirements during these tests to restrict the reactor power and reactor coolant temperature provide protection against potential conditions that could require operation of both RHR subsystems in the LPCI mode of operation. Operations in all other MODES are unaffected by this LCO.

---

ACTIONS A.1  
With one or more of the requirements of this LCO not met, (i.e., any OPERABLE IRM channel > 25/40 division of full scale on Range 7, or average reactor coolant temperature  $\geq 200^\circ\text{F}$ ) the reactor may be in a condition that requires the full complement of ECCS subsystems, and the reactor mode switch must be immediately placed in the shutdown position. This results in a condition that does not require all RHR

(continued)

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BASES

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ACTIONS

A.1 (continued)

subsystems to be OPERABLE in the LPCI mode of operation. This action may restore compliance with the requirements of this Special Operations LCO or may result in placing the plant in either MODE 3 or MODE 4.

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SURVEILLANCE  
REQUIREMENTS

SR 3.10.10.1 and SR 3.10.10.2

Periodic verification that the THERMAL POWER and reactor coolant temperature limits of this Special Operations LCO are satisfied will ensure that the stored energy in the reactor core and reactor coolant are sufficiently low to preclude the need for all RHR subsystems to be aligned in the LPCI mode of operation. The 1 hour Frequency provides frequent checks of these LCO requirements during the training startup.

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REFERENCES

1. FSAR, Section [6.3.3.4].
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# **WOG**

**NUREG 1431**

## **AFFECTED BY CHANGE PACKAGES**

**BWR-02**

**WOG-01**

**WOG-02**

**WOG-03**

**WOG-04**

**WOG-05**

**WOG-06**

**BWOG-01**

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWR-02  
NUREG 1431

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-01  
NUREG 1431

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman



# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-02  
NUREG 1431

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-03  
NUREG 1431

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-04  
NUREG 1431

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-05  
NUREG 1431

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE WOG-06  
NUREG 1431

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

# STS CHANGE CERTIFICATION FORM

CHANGE PACKAGE BWOG-01  
NUREG 1431

I certify that the changes included in this package have been accurately made in accordance with the originally approved change and are ready for public distribution.

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Technical Specification  
Subcommittee Chairman

The changes included in this package have NOT been accurately made in accordance with the originally approved change. Corrections are needed as indicated below.

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Technical Specification  
Subcommittee Chairman

1.0 USE AND APPLICATION

1.1 Definitions

-----NOTE-----

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

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

(continued)

1.1 Definitions (continued)

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CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.9.1.6. Plant operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in [Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page

(continued)



1.1 Definitions

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DOSE EQUIVALENT 1-131  
(continued)

192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity"].

$\bar{E}$  - AVERAGE  
DISINTEGRATION ENERGY

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

ENGINEERED SAFETY  
FEATURE (ESF) RESPONSE  
TIME

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

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

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

(continued)

1.1 Definitions

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LEAKAGE (continued)	<p>3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;</p> <p>b. <u>Unidentified LEAKAGE</u></p> <p>All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;</p> <p>c. <u>Pressure Boundary LEAKAGE</u></p> <p>LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.</p>
MASTER RELAY TEST	<p>A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.</p>
MODE	<p>A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.</p>
OPERABLE - OPERABILITY	<p>A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).</p>
PHYSICS TESTS	<p>PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:</p>

(continued)

1.1 Definitions

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PHYSICS TESTS  
(continued)

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

PRESSURE AND  
TEMPERATURE LIMITS  
REPORT (PTLR)

The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.9.1.7. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

QUADRANT POWER TILT  
RATIO (QPTR)

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

RATED THERMAL POWER  
(RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of [2893] MWt.

REACTOR TRIP  
SYSTEM (RTS) RESPONSE  
TIME

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

SHUTDOWN MARGIN (SDM)

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

(continued)

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1.1 Definitions

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SHUTDOWN MARGIN (SDM)  
(continued)

- a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn; and
- b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the [nominal zero power design level].

With any RCCAs not capable of being fully inserted, the reactivity worth of the RCCAs must be accounted for in the determination of SDM.

SLAVE RELAY TEST

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

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during  $n$  Surveillance Frequency intervals, where  $n$  is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER

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

TRIP ACTUATING DEVICE  
OPERATIONAL TEST  
(TADOT)

A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and-trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

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## 1.0 USE AND APPLICATION

### 1.2 Logical Connectors

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**PURPOSE** The purpose of this section is to explain the meaning of logical connectors.

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

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

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

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**EXAMPLES** The following examples illustrate the use of logical connectors.

(continued)

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## 1.0 USE AND APPLICATION

### 1.3 Completion Times

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

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

---

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

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

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

(continued)

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1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-2

ACTIONS

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

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

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

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

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for

(continued)

1.3 Completion Times

EXAMPLES  
(continued)

EXAMPLE 1.3-4

ACTIONS

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

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

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

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

(continued)



### 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

---

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

---

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

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

---

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

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

Exceptions to this Specification are stated in the individual Specifications.

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

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

---

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This

(continued)

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### 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

---

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

---

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

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

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

Exceptions to this Specification are stated in the individual Specifications.

---

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

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

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be

(continued)

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3.0 SR APPLICABILITY

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SR 3.0.3 declared not met, and the applicable Condition(s) must be  
(continued) entered.

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SR 3.0.4 Entry into a MODE or other specified condition in the  
Applicability of an LCO shall not be made unless the LCO's  
Surveillances have been met within their specified  
Frequency. This provision shall not prevent passage through  
or to MODES or other specified conditions in compliance with  
Required Actions.

---

---

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.4.1    Verify MTC is within upper limit.	Once prior to entering MODE 1 after each refueling
SR 3.1.4.2    Verify MTC is within 300 ppm Surveillance limit specified in the COLR.	-----NOTE----- Not required to be performed until 7 effective full power days (EFPD) after reaching the equivalent of an equilibrium RTP all rods out (ARO) boron concentration of 300 ppm ----- Once each cycle
SR 3.1.4.3    -----NOTES----- 1.    If the MTC is more negative than the 300 ppm Surveillance limit (not LCO limit) specified in the COLR, SR 3.1.4.3 shall be repeated once per 14 EFPD during the remainder of the fuel cycle.  2.    SR 3.1.4.3 need not be repeated if the MTC measured at the equivalent of equilibrium RTP-ARO boron concentration of $\leq 60$ ppm is less negative than the 60 ppm Surveillance limit specified in the COLR. ----- Verify MTC is within lower limit.	-----NOTE----- Not required to be performed until 7 EFPD after reaching the equivalent of an equilibrium RTP-ARO boron concentration of 300 ppm ----- Once each cycle

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ( $F_{\Delta H}^N$ )

LCO 3.2.2  $F_{\Delta H}^N$  shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. -----NOTE----- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ----- <math>F_{\Delta H}^N</math> not within limit.</p>	A.1.1 Restore $F_{\Delta H}^N$ to within limit.	4 hours
	<u>OR</u>	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	<u>AND</u>	
	A.1.2.2 Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP.	8 hours
	<u>AND</u>	
	A.2 Perform SR 3.2.2.1.	24 hours
	<u>AND</u>	
		(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be  $\leq 1.02$ .

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 Reduce THERMAL POWER $\geq 3\%$ from RTP for each 1% of QPTR > 1.00.	2 hours
	<u>AND</u>	
	A.2 Perform SR 3.2.4.1 and reduce THERMAL POWER $\geq 3\%$ from RTP for each 1% of QPTR > 1.00.	Once per 12 hours
	<u>AND</u>	
	A.3 Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours
	<u>AND</u>	Once per 7 days thereafter
	<u>AND</u>	
	A.4 Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.5 -----NOTE----- Perform Required Action A.5 only after Required Action A.4 is completed. -----</p> <p>Calibrate excore detectors to show zero QPTR.</p> <p><u>AND</u></p> <p>A.6 -----NOTE----- Perform Required Action A.6 only after Required Action A.5 is completed. -----</p> <p>Perform SR 3.2.1.1 and SR 3.2.2.2.</p>	<p>Prior to increasing THERMAL POWER above the limit of Required Action A.1</p> <p>Within 24 hours after reaching RTP</p> <p><u>OR</u></p> <p>Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1</p>
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to $\leq$ 50% RTP.	4 hours

3.6 CONTAINMENT SYSTEMS

3.6.6A Containment Spray and Cooling Systems (Atmospheric and Dual)  
(Credit taken for iodine removal by the Containment Spray System)

LCO 3.6.6A Two containment spray trains and [two] containment cooling trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours  84 hours
C. One [required] containment cooling train inoperable.	C.1 Restore [required] containment cooling train to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)



Containment Spray and Cooling Systems (Atmospheric and Dual)  
3.6.6A

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two [required] containment cooling trains inoperable.	D.1 Restore one [required] containment cooling train to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3.	6 hours
	<u>AND</u> E.2 Be in MODE 5.	36 hours
F. Two containment spray trains inoperable.  <u>OR</u> Any combination of three or more trains inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.6A.1 Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

(continued)

5.0 ADMINISTRATIVE CONTROLS

5.6 Technical Specifications (TS) Bases Control

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- 5.6.1 Changes to the Bases of the TS shall be made under appropriate administrative controls and reviewed according to Specification 5.5.1.
- 5.6.2 Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:
- a. A change in the TS incorporated in the license; or
  - b. A change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.
- 5.6.3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- 5.6.4 Proposed changes that meet the criteria of Specification 5.6.2.a or Specification 5.6.2.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71.
- 
-

5.0 ADMINISTRATIVE CONTROLS

5.7 Procedures, Programs, and Manuals

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5.7.1 Procedures

5.7.1.1 Scope

Written procedures shall be established, implemented, and maintained covering the following activities:

- a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;
- b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in [Generic Letter 82-33];
- c. Security plan implementation;
- d. Emergency plan implementation;
- e. Quality assurance for effluent and environmental monitoring;
- f. Fire Protection Program implementation; and
- g. All programs specified in Specification 5.7.2.

5.7.1.2 Review and Approval

Each procedure of Specification 5.7.1.1, and changes thereto, shall be reviewed in accordance with Specification 5.5.1, approved by the [Plant Superintendent] or his designee in accordance with approved administrative procedures prior to implementation, except as specified in Specification 5.7.1.3, and reviewed periodically as set forth in administrative procedures.

5.7.1.3 Temporary Changes

Temporary changes to procedures of Specification 5.7.1.1 may be made provided:

- a. The intent of the existing procedure is not altered;
- b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator license on the unit affected; and

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.2 Process Control Program (PCP) (continued)

- b. Shall be effective after review and acceptance by the [review method of Specification 5.5.1] and the approval of the [Plant Superintendent].

5.7.2.3 Offsite Dose Calculation Manual (ODCM)

- a. The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program; and
- b. The ODCM shall also contain the Radioactive Effluent Controls and Radiological Environmental Monitoring programs required by Specification 5.7.2.7 and Specification 5.7.2.8, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Specification [5.9.1.3] and Specification [5.9.1.4].

Licensee initiated changes to the ODCM:

- a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
  - 1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and
  - 2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;
- b. Shall become effective after review and acceptance by the [review method of Specification 5.5.1] and the approval of the [Plant Superintendent]; and

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.13 Steam Generator (SG) Tube Surveillance Program (continued)

- b. The establishment of SG tube inspection frequency dependent upon inspection result classification. Inspection frequency shall be in accordance with [Regulatory Guide 1.83, Revision [ ], date].
- c. SG tube plugging/repair limits. These limits shall be [40]% of the nominal tube wall thickness consistent with [Regulatory Guide 1.83, Revision [ ], date].
- d. Specific definitions and limits for SG tube inservice inspection acceptance criteria consistent with [Regulatory Guide 1.83, Revision [ ], date].

The content and frequency of written reports shall be in accordance with Specification 5.9.2.

The provisions of SR 3.0.2 are applicable to SG Tube Surveillance Program inspection frequencies except those established by Category C-3 inspection results.

[Key elements to be discussed and provided.]

5.7.2.14 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and

(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.14 Secondary Water Chemistry Program (continued)

- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.7.2.15 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in [Regulatory Guide ], and in accordance with [Regulatory Guide 1.52, Revision 2, ASME N510-1989, and AG-1].

- a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [ $\pm 10\%$ ].

ESF Ventilation System

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Flowrate

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- b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < [0.05]% when tested in accordance with [Regulatory Guide 1.52, Revision 2, and ASME N510-1989] at the system flowrate specified below [ $\pm 10\%$ ].

ESF Ventilation System

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Flowrate

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(continued)

5.7 Procedures, Programs, and Manuals

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5.7.2.16 Explosive Gas and Storage Tank Radioactivity Monitoring Program  
(continued)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.7.2.17 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
  1. an API gravity or an absolute specific gravity within limits,
  2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
  3. a clear and bright appearance with proper color;
- b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and
- c. Total particulate concentration of the fuel oil is  $\leq 10$  mg/l when tested every 31 days in accordance with ASTM D-2276, Method A-2 or A-3.

5.7.2.18 Fire Protection Program

This program provides controls to ensure that appropriate fire protection measures are maintained to protect the plant from fire and to ensure the capability to achieve and maintain safe shutdown in the event of a fire is maintained.

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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

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SAFETY LIMITS

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

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

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

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APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4,

(continued)

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BASES

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APPLICABILITY (continued) 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

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SAFETY LIMIT VIOLATIONS The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

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

2.2.3

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

2.2.4

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

2.2.5

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

(continued)

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BASES

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SAFETY LIMIT  
VIOLATIONS

2.2.5 (continued)

reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.6

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

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, 1988.
  2. FSAR, Section [7.2].
  3. WCAP-8746-A, March 1977.
  4. WCAP-9273-NP-A, July 1985.
  5. 10 CFR 50.72.
  6. 10 CFR 50.73.
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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

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

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

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

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

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SAFETY LIMITS

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

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(continued)

BASES

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SAFETY LIMIT  
VIOLATIONS  
(continued)

2.2.3

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

2.2.4

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

2.2.5

If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be provided to the Plant Superintendent and the Vice President-Nuclear Operations, and the [offsite reviewers specified in Specification 5.5.2] ["Offsite Review and Audit"].

2.2.6

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

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.

(continued)

BASES

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REFERENCES  
(continued)

4. 10 CFR 100.
  5. FSAR, Section [7.2].
  6. USAS B31.1, Standard Code for Pressure Piping,  
American Society of Mechanical Engineers, 1967.
  7. 10 CFR 50.72.
  8. 10 CFR 50.73.
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BASES

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SR 3.0.2  
(continued)

Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per ..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

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SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the

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BASES

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SR 3.0.3  
(continued)

required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

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SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure

(continued)

BASES

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SR 3.0.4  
(continued)

safe operation of the unit. This Specification applies to changes in MODES or other specified conditions in the Applicability associated with unit shutdown as well as startup. The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs' annotation is found in Section 1.4, Frequency.

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

- a. Inadvertent boron dilution;
- b. An uncontrolled rod withdrawal from subcritical or low power condition;
- c. Startup of an inactive reactor coolant pump (RCP); and
- d. Rod ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level trip or a high pressurizer pressure trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The startup of an inactive RCP will not result in a "cold water" criticality, even if the maximum difference in temperature exists between the SG and the core. The maximum positive reactivity addition that can occur due to an inadvertent RCP start is less than half the minimum required SDM. An idle RCP cannot, therefore, produce a return to power from the hot standby condition.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of the NRC Policy Statement. Even though it is not directly observed from the control room, SDM is considered an initial condition process variable because it is periodically monitored to ensure that the unit is operating within the bounds of accident analysis assumptions.

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LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

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APPLICABILITY

In MODE 2 with  $k_{eff} < 1.0$  and in MODES 3 and 4, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. [In MODE 5, SDM is addressed by LCO 3.1.2, "SHUTDOWN MARGIN (SDM) -  $T_{avg} \leq 200^{\circ}\text{F}$ ."] In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7.

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ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that

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BASES

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## ACTIONS

A.1 (continued)

boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank, or the borated water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1%  $\Delta k/k$  must be recovered and a boration flow rate of [ ] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by 1%  $\Delta k/k$ . These boration parameters of [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example.

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SURVEILLANCE  
REQUIREMENTSSR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.6 and LCO 3.1.7 are met. In the event that a rod is known to be untrippable, however, SDM verification must account for the worth of the untrippable rod as well as another rod of maximum worth.

In MODES 3, 4, and 5, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

- a. RCS boron concentration;
- b. Control bank position;

(continued)

BASES (continued)

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LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The boron dilution accident (Ref. 2) is the most limiting analysis that establishes the SDM value of the LCO. For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

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APPLICABILITY

In MODE 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 3 and 4, the SDM requirements are given in LCO 3.1.1, "SHUTDOWN MARGIN (SDM) -  $T_{avg} > 200^{\circ}\text{F}$ ." In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration." In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.6, "Shutdown Bank Insertion Limits," and LCO 3.1.7.

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ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid storage tank or the borated water storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may

(continued)

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 5).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factor ( $F_Q(Z)$ ) and the nuclear enthalpy hot channel factor ( $F_{\Delta H}^N$ ) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and  $F_Q(Z)$  and  $F_{\Delta H}^N$  must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of  $F_Q(Z)$  and  $F_{\Delta H}^N$  to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Policy Statement.

---

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The OPERABILITY requirements also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed.

(continued)

BASES

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BACKGROUND  
(continued)

Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

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APPLICABLE  
SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.7, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) -  $T_{avg} > 200^{\circ}\text{F}$ ," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) -  $T_{avg} \leq 200^{\circ}\text{F}$ ") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

- a. There be no violations of:

(continued)

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BASES (continued)

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ACTIONS

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the effects listed in SR 3.1.1.1.

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.6.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.6.1 (continued)

position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
  2. 10 CFR 50.46.
  3. FSAR, Chapter [15].
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BASES

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ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2 (continued)

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

C.1

If Required Actions A.1 and A.2, or B.1 and B.2 cannot be completed within the associated Completion Times, the plant must be brought to MODE 3, where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.7.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.7.1 (continued)

flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.7.2

With an OPERABLE bank insertion limit monitor, verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours. If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

SR 3.1.7.3

When control banks are maintained within their insertion limits as checked by SR 3.1.7.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.7.2.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 and GDC 26.
  2. 10 CFR 50.46.
  3. FSAR, Chapter [15].
  4. FSAR, Chapter [15].
  5. FSAR, Chapter [15].
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BASES

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BACKGROUND  
(continued)

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called group step counters) and the [Digital] Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise ( $\pm 1$  step or  $\pm \frac{5}{8}$  inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position, but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which is 6 steps. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one system fails, the DRPI will go on half accuracy with an effective coil spacing of 7.5 inches, which is 12 steps. Therefore, the normal indication accuracy of the DRPI System is  $\pm 6$  steps ( $\pm 3.75$  inches), and the maximum uncertainty is  $\pm 12$  steps ( $\pm 7.5$  inches). With an indicated deviation of 12 steps between the group step counter and DRPI, the maximum deviation between actual rod position and the demand position could be 24 steps, or 15 inches.

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APPLICABLE  
SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.6, "Shutdown Bank Insertion Limits," and

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

LCO 3.1.7, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.5, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of the NRC Policy Statement. The control rod position indicators monitor control rod position, which is an initial condition of the accident.

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LCO

LCO 3.1.8 specifies that one DRPI System and one Bank Demand Position Indication System be OPERABLE for each control rod. For the control rod position indicators to be OPERABLE requires meeting the SR of the LCO and the following:

- a. The DRPI System indicates within 12 steps of the group step counter demand position as required by LCO 3.1.5, "Rod Group Alignment Limits";
- b. For the DRPI System there are no failed coils; and
- c. The Bank Demand Indication System has been calibrated either in the fully inserted position or to the DRPI System.

The agreement between the Bank Demand Position Indication System and the DRPI System is within the limit, indicating that the Bank Demand Position Indication System is adequately calibrated for measurement of control rod bank position.

A deviation of less than the allowable limit, given in the COLR, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

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BASES

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LCO  
(continued)            OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

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APPLICABILITY        The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.5, LCO 3.1.6, and LCO 3.1.7), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

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ACTIONS              The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1

When one DRPI channel per group fails, the position of the rod can still be determined by use of the incore movable detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.1.9.2

Verification of the Power Range Neutron Flux-High trip setpoints within 8 hours prior to initiation of the PHYSICS TESTS will ensure that the RTS is properly set to perform PHYSICS TESTS.

SR 3.1.9.3

The performance of SR 3.2.1.1 and SR 3.2.2.1 measures the core  $F_0(Z)$  and the  $F_{\Delta H}^N$ , respectively. If the requirements of these LCOs are met, the core has adequate protection from exceeding its design limits, while other LCO requirements are suspended. The Frequency of 12 hours is based on operating experience and the practical amount of time that it may take to run an incore flux map and calculate the hot channel factors.

SR 3.1.9.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

- a. Reactor Coolant System (RCS) boron concentration;
- b. Control bank position;
- c. RCS average temperature;
- d. Fuel burnup based on gross thermal energy generation;
- e. Xenon concentration; and
- f. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in the calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS. The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident without the required SDM.

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(continued)

BASES (continued)

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- REFERENCES
1. 10 CFR 50, Appendix B, Section XI, 1988.
  2. 10 CFR 50.59.
  3. Regulatory Guide 1.68, Revision 2, August 1978.
  4. ANSI/ANS-19.6.1-1985, December 13, 1985.
  5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report", July 1985.
  6. FSAR, Section [14].
  7. WCAP-11618, November 1987, and Addendum 1, April 1989.
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BASES

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BACKGROUND  
(continued)

all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

- a. Critical Boron Concentration - Control Rods Withdrawn;
- b. Critical Boron Concentration - Control Rods Inserted;
- c. Control Rod Worth;
- d. Isothermal Temperature Coefficient (ITC); and
- e. Neutron Flux Symmetry.

The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

- a. The Critical Boron Concentration - Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical ( $k_{eff} = 1.0$ ), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.
- b. The Critical Boron Concentration - Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1%  $\Delta k/k$  when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron

(continued)



BASES

BACKGROUND  
(continued)

concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.5, "Rod Group Alignment Limits"; LCO 3.1.6, "Shutdown Bank Insertion Limit"; or LCO 3.1.7, "Control Bank Insertion Limits."

- c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has three alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control banks. Performance of this test could violate LCO 3.1.5, LCO 3.1.6, or LCO 3.1.7.
- d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of

(continues)

BASES

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APPLICABLE  
 SAFETY ANALYSES  
 (continued)

problems, may require the operating control or process variables to deviate from their LCO limitations.

The FSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Tables [14.1-1 and 14.1-2] summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.4, "Moderator Temperature Coefficient (MTC)," LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to  $\leq 5\%$  RTP, the reactor coolant temperature is kept  $\geq 531^\circ\text{F}$ , and SDM is  $\geq [1.6]\% \Delta k/k$ .

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR, which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. PHYSICS TESTS meet the criteria for inclusion in the Technical Specifications, since the components and process variable LCOs suspended during PHYSICS TESTS meet Criteria 1, 2, and 3 of the NRC Policy Statement.

Reference 6 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

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LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. Operation beyond specified

(continued)

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BASES

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ACTIONS  
 (continued)

C.1 and C.2

When the RCS lowest  $T_{avg}$  is  $< 531^{\circ}\text{F}$ , the appropriate action is to restore  $T_{avg}$  to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring  $T_{avg}$  to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below  $531^{\circ}\text{F}$  could violate the assumptions for accidents analyzed in the safety analyses. If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
 REQUIREMENTS

SR 3.1.10.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.10.2

Verification that the RCS lowest loop  $T_{avg}$  is  $\geq 531^{\circ}\text{F}$  will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

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BASES

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ACTIONS  
(continued)

C.1

If Required Actions A.1 through A.4 or B.1 are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.1.1 and SR 3.2.1.2 are modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that  $F_0^C(Z)$  and  $F_0^W(Z)$  are within their specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which they were last verified to be within specified limits. Because  $F_0^C(Z)$  and  $F_0^W(Z)$  could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of  $F_0^C(Z)$  and  $F_0^W(Z)$  are made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of  $F_0^C(Z)$  and  $F_0^W(Z)$  following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of  $F_0^C(Z)$  and  $F_0^W(Z)$ . The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which  $F_0$  was last measured.

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.2.1.1

Verification that  $F_0^C(Z)$  is within its specified limits involves increasing  $F_0^M(Z)$  to allow for manufacturing tolerance and measurement uncertainties in order to obtain  $F_0^C(Z)$ . Specifically,  $F_0^M(Z)$  is the measured value of  $F_0(Z)$  obtained from incore flux map results and  $F_0^C(Z) = F_0^M(Z)$  [1.0815] (Ref. 4).  $F_0^C(Z)$  is then compared to its specified limits.

The limit with which  $F_0^C(Z)$  is compared varies inversely with power above 50% RTP and directly with a function called  $K(Z)$  provided in the COLR.

Performing this Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the  $F_0^C(Z)$  limit is met when RTP is achieved, because peaking factors generally decrease as power level is increased.

If THERMAL POWER has been increased by  $\geq 10\%$  RTP since the last determination of  $F_0^C(Z)$ , another evaluation of this factor is required [12] hours after achieving equilibrium conditions at this higher power level (to ensure that  $F_0^C(Z)$  values are being reduced sufficiently with power increase to stay within the LCO limits).

The Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup because such changes are slow and well controlled when the plant is operated in accordance with the Technical Specifications (TS).

SR 3.2.1.2

The nuclear design process includes calculations performed to determine that the core can be operated within the  $F_0(Z)$  limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation,  $Z$ , is called  $W(Z)$ . Multiplying the measured total peaking factor,  $F_0^C(Z)$ , by  $W(Z)$  gives the

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.1.2 (continued)

maximum F<sub>0</sub>(Z) calculated to occur in normal operation, F<sub>0</sub><sup>M</sup>(Z).

The limit with which F<sub>0</sub><sup>M</sup>(Z) is compared varies inversely with power and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. F<sub>0</sub><sup>M</sup>(Z) evaluations are not applicable for the following axial core regions, measured in percent of core height:

- a. Lower core region, from 0 to 15% inclusive; and
- b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If F<sub>0</sub><sup>M</sup>(Z) is evaluated and found to be within its limit, an evaluation of the expression below is required to account for any increase to F<sub>0</sub><sup>M</sup>(Z) that may occur and cause the F<sub>0</sub>(Z) limit to be exceeded before the next required F<sub>0</sub>(Z) evaluation.

If the two most recent F<sub>0</sub>(Z) evaluations show an increase in the expression

$$\text{maximum over } z \quad \left[ \frac{F_0^C(Z)}{K(Z)} \right],$$

it is required to meet the F<sub>0</sub>(Z) limit with the last F<sub>0</sub><sup>M</sup>(Z) increased by a factor of [1.02], or to evaluate F<sub>0</sub>(Z) more frequently, each 7 EFPD. These alternative requirements prevent F<sub>0</sub>(Z) from exceeding its limit for any significant period of time without detection.

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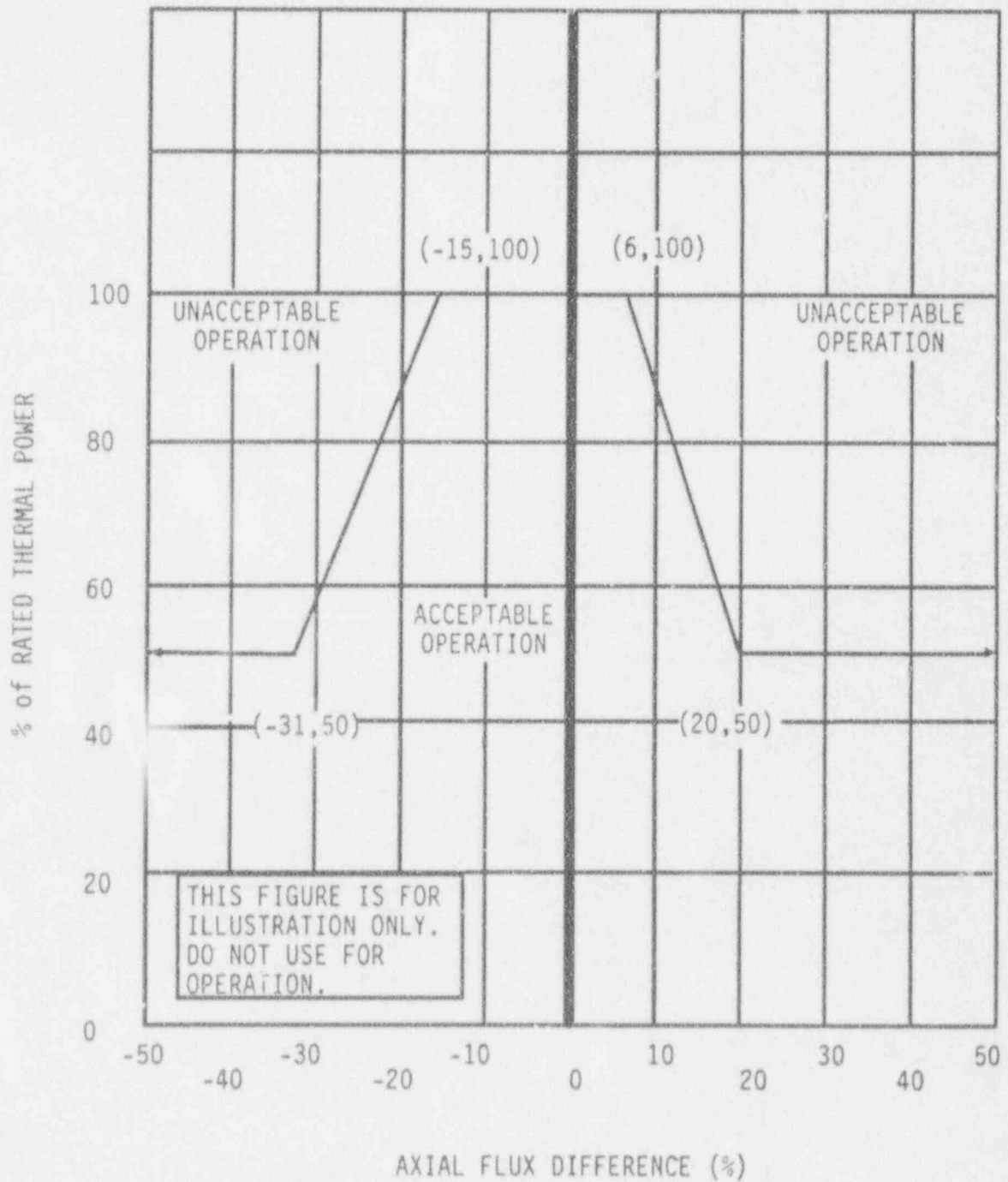


Figure B 3.2.3B-1 (page 1 of 1)  
 AXIAL FLUX DIFFERENCE Acceptable Operation Limits  
 as a Function of RATED THERMAL POWER

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

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BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.7, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

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APPLICABLE  
SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

- a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200° F (Ref. 1);
- b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition;
- c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 2); and
- d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ( $F_0(Z)$ ), the Nuclear Enthalpy Rise Hot

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

Channel Factor ( $F_{\Delta_H}^N$ ), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that  $F_{\Delta_H}^N$  and  $F_0(Z)$  remain below their limiting values by preventing an undetected change in the gross radial power distribution.

In MODE 1, the  $F_{\Delta_H}^N$  and  $F_0(Z)$  limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses.

The QPTR satisfies Criterion 2 of the NRC Policy Statement.

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LCO

The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in  $F_0(Z)$  and ( $F_{\Delta_H}^N$ ) is possibly challenged.

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APPLICABILITY

The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits.

Applicability in MODE 1  $\leq$  50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the  $F_{\Delta_H}^N$  and  $F_0(Z)$  LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.

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ACTIONS

A.1

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient

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BASES

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ACTIONS

A.1 (continued)

time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition.

A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. If the QPTR continues to increase, THERMAL POWER has to be reduced accordingly. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors  $F_{\Delta_n}^M$  and  $F_0(Z)$  are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on  $F_{\Delta_n}^M$  and  $F_0(Z)$  within the Completion Time of 24 hours ensures that these primary indicators of power distribution are within their respective limits. A Completion Time of 24 hours takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate  $F_{\Delta_n}^M$  and  $F_0(Z)$  with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although  $F_{\Delta_n}^M$  and  $F_0(Z)$  are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded

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BASES

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ACTIONS

A.4 (continued)

and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution. This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are recalibrated to show a zero QPTR prior to increasing THERMAL POWER to above the limit of Required Action A.1. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by a Note that states that the QPT is not zeroed out until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). This Note is intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is zeroed out (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that  $F_0(Z)$  and  $F_{\Delta_w}^{\#}$  are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the

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BASES

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ACTIONS

A.6 (continued)

core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been calibrated to show zero tilt (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are calibrated to show zero tilt and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE  
REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by a Note that allows QPTR to be calculated with three power range channels if THERMAL POWER is < 75% RTP and one power range channel is inoperable.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days when the QPTR alarm is OPERABLE is acceptable because of the low probability that this alarm can remain inoperable without detection.

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## BASES

SURVEILLANCE  
REQUIREMENTSSR 3.2.4.1 (continued)

When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is required only when one power range channel is inoperable and the THERMAL POWER is  $\geq 75\%$  RTP.

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 for three and four loop cores.

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.2.4.2 (continued)

occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

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REFERENCES

1. 10 CFR 50.46.
  2. Regulatory Guide 1.77, Rev [0], May 1974.
  3. 10 CFR 50, Appendix A, GDC 26.
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BASES

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ACTIONS

D.1 (continued)

needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System, the iodine removal function of the Containment Spray System, and the low probability of DBA occurring during this period.

E.1 and E.2

If the Required Action and associated Completion Time of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With two containment spray trains or any combination of three or more containment spray and cooling trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.6A.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (only check valves are inside containment) and capable of potentially being mispositioned are in the correct position.

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BASES

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ACTIONS

A.3 (continued)

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

A minimum frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The frequency is based on operating experience, which has shown 72 hours to be adequate.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
  2. FSAR, Chapter [15].
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BASES

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BACKGROUND  
(continued)

must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements (Ref. 1).

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APPLICABLE  
SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.

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LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.4.1 (continued)

demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.4.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. In LCO 3.3.6, the Containment Purge and Exhaust Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

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(continued)

BASES

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APPLICABILITY  
(continued) Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

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ACTIONS RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS because all of unborated water sources are isolated.

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling water level  $\geq$  23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

(continued)

BASES

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ACTIONS  
(continued)

A.4

If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.

The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.5.1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

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REFERENCES

1. FSAR, Section [5.5.7].
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