DCL-97-106, Attachment 3
(List of Additional Changes & Related LARs)
ERRATA

DIABLO CANYON TABLE OF CHANGES NOT WITHIN THE SCOPE OF FULL CONVERSION TO THE ISTS[2]

ITS Change No./CTS Change No.	Description
	ITS 1.0/CTS 1.0
ITS 1.1-7 CTS 01-01-A CTS 01-30-A	The definition of a CHANNEL FUNCTIONAL TEST (CFT) from the current Technical Specifications (CTS) was included in the ITS. However, this definition was revised in the improved Technical Specifications (ITS) to allow an actual signal, a required actuation, or any series of overlapping tests to be credited for satisfying the requirements of the test. The same changes were made to the CHANNEL OPERATIONAL TEST (COT) via TSTF-39. A COT and a CFT test similar channel functions.
	ITS 3.3/CTS 3/4.3
ITS 3.3-104 CTS ~~02-36-M~~ 02-48-LS28	Action 15 was added to CTS 3.3.2 to describe the actions required when both first or second level 4kV undervoltage relays are inoperable. This change has been proposed in License Amendment Request (LAR) 97-02.
ITS 3.3-29 CTS 02-29-M	An Engineered Safety Features Actuation System function for ~~the~~ residual heat removal pump trip on low refueling water storage tank (RWST) level ~~channels~~ is added. This change ~~will~~ was also ~~be~~ included in a separate LAR 97-10 ~~to be~~ submitted ~~by approximately June~~ July 30, 1997.
	ITS 3.4/CTS 3/4.4
ITS 3.4-45 CTS ~~04-01-LS~~ 09-08-LS7	The low temperature overpressure protection (LTOP) system LCO note on centrifugal charging pump (CCP) swap is revised to allow both CCPs to be capable of injecting into the RCS for up to 4 hours throughout the LTOP applicability. not an out-of-scope change per WOG-51
ITS N/A CTS 05-03-A	(See ITS 5.5.9, ~~e.4.a.8,~~ d.1.h ~~and Section 3/4.0, CN 1-15-A.~~) The definition of "Tube Inspection" is clarified to eliminate potential misunderstanding with regard to the required point of entry.
	ITS 3.5/CTS 3/4.5
ITS N/A CTS 01-08-A CTS 05-03-A	The volume of the RWST and accumulators is revised to be expressed in percent level rather than gallons, as specified

[2] Changes to the ISTS except those which involve the incorporation of plant specific design information, which were developed as part of the industry traveler process, which are simple editorial corrections, or which incorporate CTS information; and changes to the CTS that do not merit a separate LAR.

	in the ITS and the CTS.
ITS 3.6/CTS 2/4.6	
ITS 3.6-10 CTS 09-01-A	The volume of the spray additive tank is revised to be expressed in percent volume rather than gallons as indicated in the ITS and the CTS.
ITS 3.6-13 CTS 07-10-LS9	A note is added to delete the surveillance requirement to leak test containment ventilation isolation valves with resilient seals if the flow path is isolated by a leak tested blank flange.
ITS 3.7/CTS 3/4.7	
ITS 3.7-01 CTS ~~01-01-A~~ 01-04-LS3	This change modifies the power range neutron flux high trip setpoints to reflect a new algorithm used to determine the setpoints. The algorithm was introduced in Westinghouse Nuclear Safety Advisory Letter 94-001. This change ~~will~~ was also ~~be~~ included in a ~~future~~, separate LAR 97-06
ITS - N/A CTS 05-02-LS11	The allowed outage time for the MSIVs is increased to 8 hours. NUREG-1431 has been revised via a traveler to provide an allowed outage time of 72 hours for the MSIVs.
ITS 3.7-15 CTS 09-01-M	Verification that a motive force is available to assure that valves in the ASW system that must be re-positioned can be repositioned is added to SR 3.7.8.1. This requirement is not in the CTS
ITS 3.8/CTS 3/4.8	
ITS 3.8-47 CTS 01-48-M	LCO 3.8.3, ACTION B. regarding stored diesel generator (DG) lube oil was changed from a per DG format to a plant wide, shared system bases similar to the diesel fuel oil supply
ITS 5.0/CTS 6.0	
ITS 5.7-1 CTS 03-11-A	Limitations in the Radioactive Effluent Controls Program, reporting requirements for the Occupational Radiation Exposure Report and the Annual Radiological Environmental Operating Report, and radiation limits for High Radiation Areas are revised to reflect the requirements of revised 10 CFR 20 (proposed specifications 5.5.4, 5.6.1, 5.6.3, and 5.7)

DCL-97-106, LAR 97-09
ITS 1.0 ERRATA

Ⓒ

1-09-A

~~TABLE 1.1~~

~~FREQUENCY NOTATION~~

NOTATION	FREQUENCY
~~S~~	~~At least once per 12 hours.~~
~~D~~	~~At least once per 24 hours.~~
~~W~~	~~At least once per 7 days.~~
~~M~~	~~At least once per 31 days.~~
~~Q~~	~~At least once per 92 days.~~
~~SA~~	~~At least once per 184 days.~~
~~R~~	~~At least once per 18 months.~~
~~R24, REFUELING INTERVAL~~	~~At least once per 24 months.~~
~~S/U~~	~~Prior to each reactor startup.~~
~~P~~	~~Completed prior to each release.~~
~~N.A.~~	~~Not applicable.~~

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

Insert

M

Insert for 1.0 Encl. 3B

1-08 A	The current TS definitions for Engineered Safety Features Response Time and Reactor Trip System Response Time would be modified. In addition, the term "measured" would be replace by "verified" to be consistent with the requirements of improved TS SR 3.3.1.16 and SR 3.3.2.10 to verify response time is within limits.	Yes	Yes	Yes	Yes
1-09 A	The current TS definition for Frequency Notation (and Table 1.1, Frequency Notation) would be deleted to be consistent with NUREG-1431. The acronyms defined in Table 1.1, Frequency Notation, are no longer used in NUREG-1431.	Yes	Yes	Yes	Yes

DCL-97-106, LAR 97-09
ITS 2.0 ERRATA

D

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour ~~and comply with the requirements of Specification 6.7~~. 01-02-A

delete redline and then strike-out

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour~~, and comply with the requirements of Specification 6.7~~. 01-02-A

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes~~, and comply with the requirements of Specification 6.7~~. 01-02-A

D

2.2 LIMITING SAFETY SYSTEM SETTINGS

02-01-A

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

remove the strike-out

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

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

ACTION:

02-06-LG

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

~~b. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirements of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.~~ 02-06-LG

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

02-01-A

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{1+\tau_4 S}{1+\tau_5 S} \le \Delta t_o\{K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 [T-T"] - f_2(\Delta I)\}$$

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

τ_4, τ_5 = Time constants utilized in the lead-lag controller for ΔT, τ_4 ~~=~~ $\le$ ≧ 0 seconds, τ_5 ~~=~~ ≦ 0 seconds $\ge$ 02-04-M

Δt_o = Indicated ΔT at RATED THERMAL POWER

K_4 = ≦ 1.072 02-04-M

K_5 = ≧ 0.0174/°F for increasing average temperature, and 0 for decreasing average temperature

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

τ_3 = Time constants utilized in the rate-lag controller for T_{avg}. τ_3 ~~=~~ ≦ 10 secs.

K_6 = ≧ 0.00145/°F for T > T", and 0 for T ≤ T"

T = Average temperature, °F 02-04-M

T" = Indicated T_{avg} at RATED THERMAL POWER, ≤576.6 °F (Unit 1) and ≤577.6 °F (Unit 2) 02-07-A

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

$f_2(\Delta I)$ = 0 for all ΔI

(11)

(E)

DESCRIPTION OF CHANGES TO TS SECTION 2.0

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

CHANGE NUMBER	NSHC	DESCRIPTION
01-01	A	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparison Table (Enclosure 3B).
01-02	A	The requirements embodied in separate administrative controls dealing with Safety Limits (SL) violations are deleted. Specifications 6.7 is deleted per CN 02-02-LS in Enclosure 3A of the Administrative Controls package.
02-01	A	The requirements of this LCO are moved to ITS LCO 3.3.1.
02-02	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-03	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-04	M	[to the time constraints and] to K_1, K_4, K_5, and K_6 Addition of the inequality signs consistent with NUREG-1431, indicates the conservative direction for these K values [and τ values]
02-05	A	This change corrects the OTΔT equation by relocating the bracket to the correct position, as described in CN 3.3-10 of the CTS 3/4.3 attachment.
02-06	LG	The requirements stipulated in ACTIONS [a and b] are moved to ITS Table 3.3.1-1, with explicit direction contained in [the ITS Background (Trip Setpoints and Allowable Values) Bases, ACTIONS Bases, and SR 3.3.1.10 ~~and~~, SR 3.3.1.11 Bases]. This change removes details more appropriately controlled outside of the TS while retaining those aspects necessary to assure the protection functions are performed if necessary.
02-07	A	This change incorporates the values for T' (Nominal T_{avg} at RATED THERMAL POWER), that were inadvertently deleted during a previous License Amendment.

, and 3.3.1.12

E

Industry Travelers Applicable to Section 2.0

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

BASES (Continued)

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~~ below which the ~~minimum~~ calculated DNBR is not less than ~~the safety analyses limit~~, the design DNBR value, 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 rise 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~~. D

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 ~~ΔT~~ reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ~~ΔT~~ reactor trips will reduce the setpoints to provide protection consistent with the reactor core ~~SLs (Refs. 3 and 4)~~.

*Remove Strike-out

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

SAFETY LIMIT VIOLATIONS

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

~~2.2.1~~

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

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

~~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)~~.

(Continued)

(B)

BASES (Continued)

SAFETY LIMIT VIOLATIONS (continued)

~~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.~~

~~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.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10 (associated with 1967 GDC 6 per FSAR Appendix 3.1A)
2. FSAR, ~~Section [7.2]~~ Chapter 7.2.
3. WCAP-8746-A, March 1977.
4. ~~WCAP-9273-NP-A, July 1985.~~ remove strike-out
5. ~~10 CFR 50.72.~~ ~~FSAR, Chapter 16.~~ remove strike-out

remove strike-out

6. ~~10 CFR 50.73.~~

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

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

remove strike-out

The design pressure of the RCS is 2485 psig ~~2500 psia~~. 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~~ were hydrostatically tested at ~~125%~~ 150% (3750) (Ref. 9) of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

remove strike-out

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs 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).

APPLICABLE SAFETY ANALYSES

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

BASES (Continued)

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~~ turbine trip without a direct reactor trip. ~~During the transient, no control actions are assumed, except that the~~ Safety valves on the secondary-~~plant~~ side are assumed to open when the steam pressure reaches the ~~secondary plant~~ safety valve settings. ~~and nominal~~ lost at the time of turbine trip main feedwater supply ~~is~~ is ~~maintained~~.

remove strike-out

remove strike-out

The Reactor Trip System ~~setpoints (Ref. 5)~~ allowable values 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.

remove strike-out

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

a. Pressurizer power operated relief valves (PORVs);

b. Steam ~~line relief valve~~ Generator Atmospheric Dump valves;

remove strike-out

c. Steam Dump System;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valve.

remove strike-out

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.

APPLICABILITY

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

BASES (Continued)

SAFETY LIMIT VIOLATIONS

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

~~2.2.2.1~~ strike-out

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

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).

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

~~2.2.2.2~~ strike-out

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

~~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.~~

~~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.~~

(B)

BASES (Continued)

insert 1 insert 2 insert 3

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.~~ retain strike-out remove strike-out

3. ASME. Boiler and Pressure Vessel Code. Section XI. ~~Article IWX 5000~~. 1977 Edition and Summer 1978 Addenda

4. 10 CFR 100.

5. ~~FSAR. Section [7.2].~~ remove strike-out

6. ~~USAS B31.1. Standard Code for Pressure Piping. American Society of Mechanical Engineers. 1967.~~ DCM S-7, 3.4.1

remove strike-out

7. ~~10 CFR 50.72.~~

8. ~~10 CFR 50.73.~~

9. ~~Westinghouse Report SD-117, "Structural Analysis of Reactor Coolant Loop/Support System for Diablo Canyon Nuclear Generating Station Unit No. 1," February 1975.~~

insert 1 : (associated with 1967 GDC 9 per FSAR Appendix 3.1A)

insert 2 : (no direct correlation to 1967 GDC; however, intent of 1971 GDC is met per FSAR Appendix 3.1A)

insert 3 : (associated with 1967 GDC 30 per FSAR Appendix 3.1A)

DCL-97-106, LAR 97-09
ITS 3.1 ERRATA

(C)

~~REACTIVITY CONTROL SYSTEMS~~

~~POSITION INDICATION SYSTEM - SHUTDOWN~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.~~ 14-01-R

~~APPLICABILITY: MODES 3*#, 4*# and 5*#.~~

~~ACTION:~~

~~With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.~~

~~SURVEILLANCE REQUIREMENTS~~

4.1.3.3 Each of the above required digital rod position indicators shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full range of rod travel at least once ~~each REFUELING INTERVAL.~~ prior to criticality after each removal of the reactor head 13-07-M

~~*With the Reactor Trip System breakers in the closed position.~~

~~#See Special Test Exceptions Specification 3.10.4~~ 14-01-R 3-01-A

C

~~REACTIVITY CONTROL SYSTEMS~~

~~ROD DROP TIME~~

15-01-R

~~LIMITING CONDITION FOR OPERATION~~

~~3.1.3.4~~ The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. Tavg greater than or equal to ~~541~~ 500°F, and 12-10-LS10

b. All reactor coolant pumps operating.

~~APPLICABILITY: MODES 1 and 2~~.

~~ACTION~~:

With the drop time ~~of any full-length rod~~ determined to exceed the above limit, restore the rod drop time to within the above limit prior to ~~proceeding to MODE 1 or 2~~. reactor criticality 12-01-A 12-20-A

~~SURVEILLANCE REQUIREMENTS~~ 12-01-A

~~4.1.3.4~~ The rod drop time ~~of full-length rods~~ shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head, 15-02-A

~~b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and~~ 12-11-TR3

~~c. At least once each REFUELING INTERVAL.~~

C

DESCRIPTION OF CHANGES TO TS SECTION 3/4.1
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
12-08	LS9	Consistent with NUREG-1431, the requirement to reduce the high neutron flux set point to ≤ 85 percent of RTP would be deleted. This is acceptable because the underlying safety limits are not of a nature that requires immediate shutdown of the plant if they are exceeded. This is evidenced by the allowance of 72 hours to verify peaking factors. It is assumed that during this 72-hour period an event will not occur which will raise the power level and cause a high neutron flux trip at 100 percent RTP. If a power excursion would occur from the 75 percent RTP ACTION statement limit, the initial peaking factors would not be critical to the analysis, since the analysis is based on the peaking factors at 100 percent RTP. Therefore, the risk of a reactor trip caused by adjusting the power range trip set points is not justified by the potential consequences of failing to reduce the trip set points.
12-09	M	Not applicable to DCPP. See Conversion Comparison Table. (Enclosure 3B)
12-10	LS10	The requirement to maintain RCS $T_{avg} \geq 541°F$ during rod drop testing would be revised to maintain $T_{avg} \geq 500°F$. NUREG-1431, allows the tests to be performed at temperatures as low as 500°F. Because the RCS coolant is more dense at lower temperatures, the rod drop time would be greater at the lower temperatures than at the higher temperatures. In addition, the RCS is borated such that the SDM remains within its limits at the Conditions existing during these tests. Nevertheless, this change, which allows more flexibility of plant conditions for conducting rod drop testing, is a relaxation in plant operations in the CTS.
12-11	TR3	24 It is proposed to move the requirement to perform rod drop testing on individual rods following maintenance that could affect the drop time to licensee controlled documents [and to delete the ~~18~~ month requirement]. The requirement to perform drop time testing following each removal of the reactor vessel head would not be modified. The proposed change is justified, because in addition to being consistent with NUREG-1431, Rev. 1, good maintenance practices would require a retest following any maintenance on a rod or its drive system that could affect drop time. Furthermore, it is difficult to postulate any maintenance on a rod that could affect its drop time without requiring the reactor vessel head to be removed in the process. The components of the rod and rod drive system that affect drop time (as defined in TS) are all inside the reactor coolant pressure boundary. Therefore, moving this requirement outside the TS would have essentially no impact on rod OPERABILITY. [Measuring rod drop time following each removal of the reactor vessel head is considered equivalent if not more restrictive than an ~~18~~ month frequency requirement; therefore, deleting the ~~18~~ month requirement where it exists (not all plants have it in the CTS) is an administrative change.] 24

TECH SPECH CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-09 M	This proposed change would reinstate an SDM verification requirement that had been eliminated in a previous license amendment.	No, already in CTS.	No, already in CTS.	Yes	Yes
12-10 LS10	The requirement to maintain RCS $T_{avg} \geq [541]°F$ during rod drop testing would be revised to maintain $T_{avg} \geq 500°F$.	Yes	Yes	Yes	Yes
12-11 TR3	The requirement to perform drop testing on rods following maintenance would be removed from the CTS.	Yes, also deletes redundant ~~18~~ 24 month interval.	Yes, also deletes redundant 18 month interval.	Yes	Yes
12-12 LS13	CTS [3.1.3.1] ACTIONS are revised to delete reference causes of control rod inoperability due to rod control urgent failure or other electrical problems in the rod control system.	Yes	Yes	Yes	No, CTS already revised to incorporate.
12-13	Not used.	N/A	N/A	N/A	N/A
12-14 M	This wording is broadened from "untrippable" to "inoperable" to ensure all causes of inoperability are covered. This more restrictive change clarifies the appropriate ACTIONS to be taken for all causes of inoperability, consistent with Traveler TSTF-107.	Yes	Yes	Yes	Yes
12-15 A	Rod misalignment is determined based on a comparison between the rod's DRPI and its group step counter demand position, not on a rod to rod position verification. This change is administrative in nature in that there is no effect on the manner in which the operating staff would determine whether a misalignment event had occurred.	No, already in CTS.	No, already in CTS.	Yes	Yes
12-16 LG	Several surveillances (e.g., rod position deviation monitor *and rod insertion limit monitor in this section)* contain ACTIONS in the form of increased surveillance frequency to be performed in the event of inoperable alarms. These ACTIONS are relocated from the TS to licensee controlled documents. This is consistent with TSTF-110, Rev. 1.	Yes, moved to the FSAR.	Yes, moved to TRM.	Yes, moved to the USAR.	Yes, moved to the FSAR, Section 16.1.

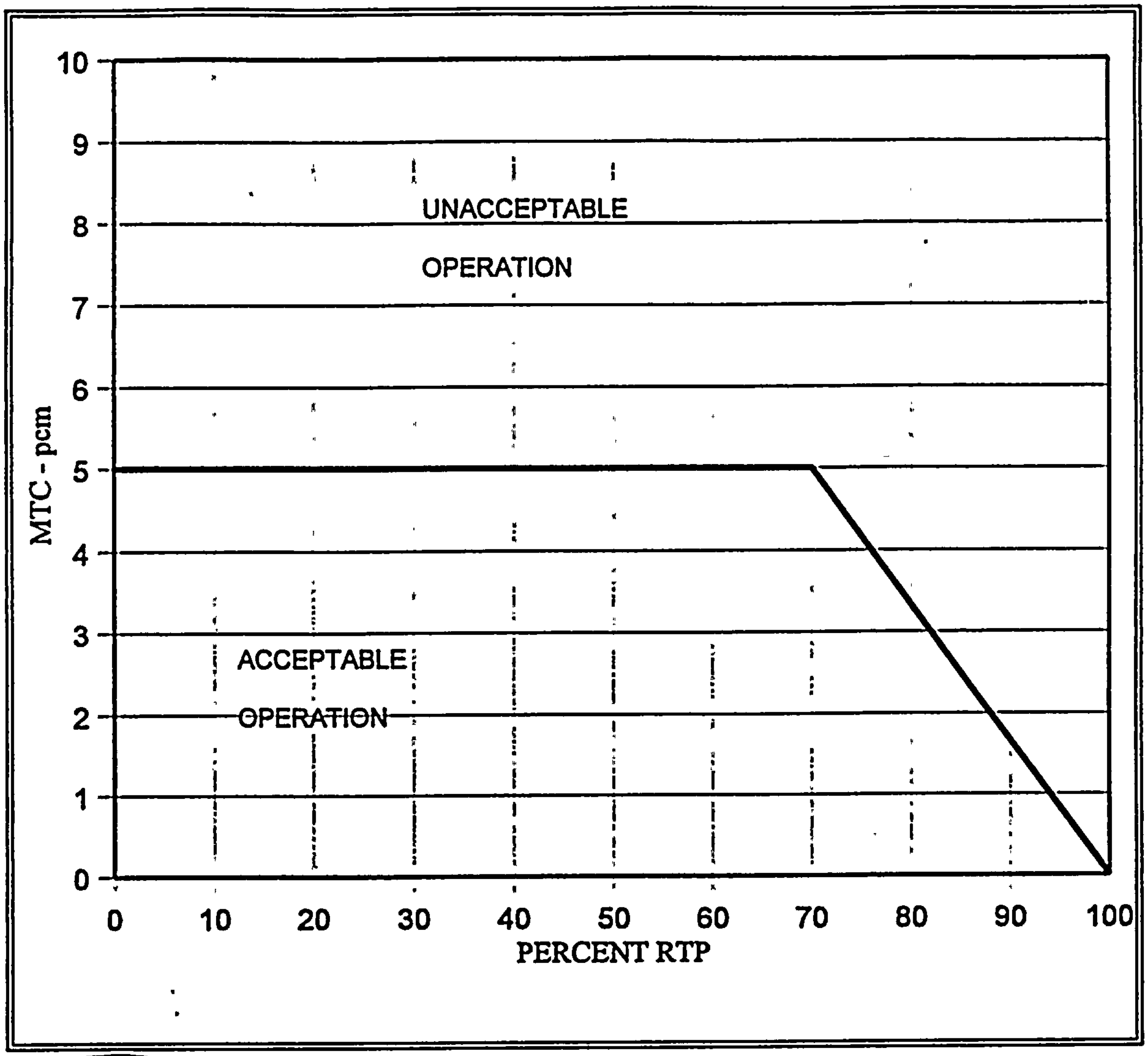

FIGURE 3.1.3-1 (page 1 of 1) MODERATOR TEMPERATURE COEFFICENT vs. POWER LEVEL

D

APPLICABLE SAFETY ANALYSIS (continued)

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 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 coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

~~will~~ "RED LINE"

The startup of an inactive RCP in MODES 1 or 2 is precluded. In MODE 3, the startup of an inactive RCP can~~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 start is less than half the minimum required X

(Continued)

X

~~In determining the boration flow rate, the time in core life must be considered. Forinstance, the most dificult time in core life to increase the RCS boron concentration is at the begining of cycle when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% Δk/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 approcimately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of the RCS~~ ~~parameters will increase the SDM by 1% Δk/k. These boration parameters of [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example.~~ X

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODES 1 and 2, SDM is verified by observing that the requirements of LCO 3.1.~~6~~5 and LCO 3.1.~~7~~6 are met. In the event that a rod is known to be untrippable, however, SDM

APPLICABLE SAFETY ANALYSIS (Ref. 2 (Continued))

The consequences of accidents that cause core overheating must be evaluated when the is positive. Such accidents include the rod withdrawal transient from either zero or RTP, loss of main feedwater flow, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and sudden decrease in feedwater temperature.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis (Ref. 2).

"Red Line"

"Red Line"

MTC values are bounded in reload safety evaluations assuming steady state conditions at BOC and EOC. An EOC measurement is conducted at conditions when the RCS boron concentration reaches a boron concentration equivalent to 300 ppm at an equilibrium, all rods out RTP condition. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions. X

The most negative MTC value, equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved: (1) a conversion of the MDC used in the FSAR accident analyses to its equivalent MTC, based on the rate of change of moderator density with temperature at RATED THERMAL POWER conditions, and (2) adding margin to this value to account for the largest difference in MTC observed between an EOC, all rods withdrawn, RATED THERMAL POWER condition and an envelope of those most adverse conditions of moderator temperature and pressure, rods inserted to their insertion limits, axial power skewing, and xenon concentration that can occur in normal operation within Technical Specification limits and lead to a significantly more negative EOC MTC at RATED THERMAL POWER. These corrections transformed the MDC value used in the FSAR accident analyses into the limiting EOC MTC value. The 300 ppm surveillance limit MTC value represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by adding an allowance for burnup and soluble boron concentration changes to the limiting EOC MTC value.

(Continued)

A

~~with a center to center distance of 3.75 inches, which is six steps.~~

BASES

~~(± 7.5 inches)~~

BACKGROUND position (continued)

group all receive the same signal to move and should, therefore, all be at the same indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ 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. To increase the reliability of the system, the inductive coils are connected alternately to data system A or B. Thus, if one data 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 ± 6 steps (±3.75 inches), and the maximum uncertainty is ±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.~~ The DRPI system is capable of monitoring rod position within at least ± 12 steps with either full accuracy or half accuracy

APPLICABLE SAFETY ANALYSIS

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing rod inoperability or misalignment are that:

a. There be no violations of:

    1. Specified acceptable fuel design limits, or

    2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control or shutdown rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

(Continued)

"RED LINE"

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 15 steps from the top of the core would require a significant power reduction, since control bank D must be fully inserted and control bank C must be inserted to approximately 100 steps ~~to 115 steps~~.

Power operation may continue with one RCCA OPERABLE (i.e. trippable) but misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, ~~RTP~~ reactor power must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible. Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 3). The Completion

R

X

X

X

(Continued)

~~D.1~~ C.1 "RED LINE"

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

~~C.1.1 AND C.1.2~~ D.1.1 and D.1.2 "RED LINE"

More than one ~~control~~ rod becoming misaligned from its group ~~average~~ demand position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to

(Continued)

D

BASES

ACTIONS and (continued)

complete the action. This allows the operator sufficient time to align the required valves start the boric acid pumps. Boration will continue until the required SDM is restored.

Additionally, the requirements of LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," apply if the misaligned rods are not within the required insertion limits.

~~C.2~~ D.2 "REALINE"

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant system.

SURVEILLANCE REQUIREMENTS

SR ~~3.1.5.1~~ 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. ~~If the rod position deviation monitopr is inoperable, a Frequency of 4 hours accomplishes the same goal.~~ The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

(Continued)

ACTIONS (continued)

OPERABLE and the most withdrawn rod and the least withdrawn rod are ≤ 12 steps apartwithin the allowed Completion Time of once every 8 hours is adequate.

~~C.2~~D.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors ~~limits~~ (Ref. 3). The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions ~~C.1.1~~ D.1.1 and ~~C.1.2~~ D.1.2 or reduce power to ≤ 50% RTP.

~~D.1~~E.1

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 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.~~8~~7.1

Verification that the DRPI agrees with the demand position within 12 steps ensures that the DRPI is operating correctly. Verification at 24, 48, 120, and 228 steps withdrawn for the control and shutdown banks provides assurance that the DRPI is operating correctly over the full range of indication. ~~Since the DRPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.~~

~~The 18 month frequency is based on the need to preform~~ This surveillance ~~under conditions that apply during a plant outage and the~~ is performed prior to reactor criticality after each removal of the reactor head, since there is potential for unnecessary plant transients if the SR were performed with the reactor at power.~~.~~ ~~Operating experience has shown these components usually pass the SR when performed at a Frequency of once every 18 months. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

(Continued)

DCL-97-106, LAR 97-09
ITS 3.3 ERRATA

(C)

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE ~~with RESPONSE TIMES as shown in Table 3.3-2~~. 01-35-LG

APPLICABILITY: As shown in Table 3.3-1.

ACTION:* 01-01-A

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The required ~~REACTOR~~ TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be ~~demonstrated~~ verified ~~to be~~ within its limit at least once per 24 months on a STAGGERED TEST BASIS. ~~Each test shall include at least one train such that both trains are tested at least once per 48 months and one channel per function such that all channels are tested at least once every N times 24 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.~~ 01-02-LG 01-03-LS1

(new) * Separate Condition entry allowed for each function. 01-01-A

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	~~TOTAL NO.~~ OF CHANNELS	REQUIRED ~~TO TRIP~~	~~MINIMUM~~ CHANNELS ~~OPERABLE~~	CHANNELS MODES	APPLICABLE ACTION	01-04-LG 01-43-A
1. Manual Reactor Trip	2	~~1~~	~~2~~	1, 2	1	
	2	~~1~~	~~2~~	3*, 4*, 5*	11	
2. Power Range, Neutron Flux						
a. High Setpoint	4	~~2~~	~~3~~	1, 2	2	
b. Low Setpoint	4	~~2~~	~~3~~	1###, 2	~~2~~ 2.1	01-06-LS2
3. Power Range, Neutron Flux High Positive Rate	4	~~2~~	~~3~~	1, 2	~~2~~ 2.1	01-06-LS2
4. Power Range, Neutron Flux High Negative Rate	4	~~2~~	~~3~~	1, 2	~~2~~ 2.1	01-06-LS2
5. Intermediate Range, Neutron Flux	2	~~1~~	~~2~~	1###, 2(d)	3, 3.1	01-07-LS3
6. Source Range, Neutron Flux						
a. Startup	2	~~1~~	~~2~~	2##	4, 4.1	01-08-M
b. Shutdown	2	~~1~~	~~2~~	3*(f), 4*(f), 5*	11, 4.1	01-08-M
c. Shutdown	2	~~0~~	~~1~~	3(f), 4(f), and 5(f)	5	01-47-A
7. Overtemperature ΔT	4	~~2~~	~~3~~	1, 2	~~6~~ 2.1	01-45-M
8. Overpower ΔT	4	~~2~~	~~3~~	1, 2	~~6~~ 2.1	01-45-M
9. Pressurizer Pressure-Low	4	~~2~~	~~3~~	1(g)	6	01-19-LS8
10. Pressurizer Pressure-High	4	~~2~~	~~3~~	1, 2	~~6~~ 2.1	01-45-M
11. Pressurizer Water Level-High	3	~~2~~	~~2~~	1(g)	6	01-19-LS8

(71)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	~~TOTAL NO.~~ OF CHANNELS	~~REQUIRED TO TRIP~~	~~MINIMUM CHANNELS OPERABLE~~	CHANNELS MODES	APPLICABLE ACTION	
						01-04-LG
						01-43-A
12. Reactor Coolant Flow-Low	3/loop			1(9)	6	01-19-LS8
~~a. Single Loop (Above P-8)~~	~~3/loop~~	~~2/loop in one loop~~	~~2/loop in each loop~~	~~1~~	6	01-57-LG
~~b. Two Loops (Above P-7 and below P-8)~~	~~3/loop~~	~~2/loop in two loops~~	~~2/loop in each loop~~	~~1~~	6	
13. Steam Generator Water Level Low-Low						
a. Steam Generator Water Level-Low-Low	3/S.G.	~~2/S.G. in one S.G.~~	~~2/S.G. in each S.G.~~	1, 2	~~6~~ 21	01-45-M
b. RCS Loop ΔT	4 (1/loop)	~~N.A.~~	~~N.A.~~	1, 2	27	
14. DELETED						
15. Undervoltage-Reactor Coolant Pumps	2/bus	~~1/bus both busses~~	~~1/bus~~	1(9)	~~28~~ 6	01-19-LS8 01-50-A
16. Underfrequency-Reactor Coolant Pumps	3/bus	~~2 on same bus~~	~~2/bus~~	1(9)	~~28~~ 6	01-19-LS8 01-50-A
17. Turbine Trip						
a. Low Autostop Oil Pressure	3	~~2~~	~~2~~	1([illegible])	7	01-48-LS4
b. Turbine Stop Valve Closure	4	4	4	1([illegible])	7	01-48-LS4

(M)

~~Unit 1 — Amendment No. 103~~
~~Unit 2 — Amendment No. 102~~
~~July 2, 1995~~

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	~~TOTAL NO.~~ OF CHANNELS	REQUIRED ~~TO TRIP~~	~~MINIMUM CHANNELS OPERABLE~~	~~CHANNELS~~ APPLICABLE MODES	ACTION	
						01-04-LG 01-43-A
18. Safety Injection Input from ESF	2	~~1~~	~~2~~	1, 2	26	
19. Reactor Coolant Pump Breaker Position Trip above P-7	1/breaker	~~2~~	~~1/breaker~~	1(g)	9, 6	01-49-LS18
20. Reactor Trip Breakers (k)	2 2	~~1~~ ~~1~~	~~2~~ ~~2~~	1, 2 3*, 4*, 5*	10, ~~12~~ 11	01-14-A
(new) Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1 each per RTB 1 each per RTB			1, 2 3*, 4*, 5*	12 11	01-14-A 01-14-A
21. Automatic Trip and Interlock Logic	2 2	~~1~~ ~~1~~	~~2~~ ~~2~~	1, 2 3*, 4*, 5*	26 11	
22. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2	~~1~~	~~2~~	2##	8	01-51-LG
b. Low Power Reactor Trips Block, P-7 ~~P-10 Input~~ ~~P-13 Input~~	4 1 per train ~~2~~	~~2~~ ~~1~~	~~3~~ ~~2~~	1 ~~1~~	8# 8.1 8#	01-12-M 01-05-A 01-12-M
c. Power Range Neutron Flux, P-8	43	~~2~~	~~3~~	1	8# 8.1	01-37-A 01-05-A
d. Power Range Neutron Flux, P-9	43	~~2~~	~~3~~	1	8# 8.1	01-37-A 01-12-M
e. Power Range Neutron Flux, P-10	43	~~2~~	~~3~~	1, 2	8#	01-05-A 01-37-A
f. Turbine Impulse Chamber Pressure, P-13 (Input to P-7)	2	~~1~~	~~2~~	1	8# 8.1	01-05-A 01-05-A
23. Seismic Trip	3 directions (x,y,z) in 3 locations	~~2/3 locations one direction~~	~~2/3 locations all directions~~	1,2	13	01-12-M

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

* When the ~~Reactor Trip System breakers are in the closed position and the~~ Control Rod Drive System is capable of rod withdrawal or all rods are not fully inserted. 01-55-LS39

~~#The provisions of Specification 3.0.4 are not applicable.~~ 01-05-A

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(new) *** Not required to be performed until 12 hours after input from one Power range Neutron Flux channel to QPTR is inoperable and THERMAL POWER is > 75% RTP. 01-53-A

(new) (f) With the RTB's open or the Rod Control System incapable of withdrawal. In this condition, source range function does not provide reactor trip but does provide indication. 01-47-A

(new) (j) Above the P-9 (Power Range Neutron Flux) Interlock. 01-48-LS4

(new) (k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB. 01-14-A

(new) (d) Above the P-6 (Intermediate Range Neutron Flux) Interlock. 01-07-LS3

(new) (g) Above the P-7 (Low Power Reactor Trips Block) Interlock. 01-19-LS8

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than the ~~Minimum~~ REQUIRED Channels ~~OPERABLE requirement,~~ restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours. 01-04-LG

ACTION 2 - With the number of OPERABLE channels one less than the ~~Total Number of~~ REQUIRED Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: 01-43-A

a. The inoperable channel is placed in the tripped condition within 6 hours.

b. ~~The Minimum Channels OPERABLE requirement is met; however,~~ The inoperable channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1 and set point adjustment, and 01-04-LG 01-17-A

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER ~~and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER~~ within 4 12 hours; or, if the power range neutron flux input to the QUADRANT POWER TILT RATIO is inoperable, the QPTR is monitored per Specification 4.2.4.2*** ~~when THERMAL POWER is greater than or equal to 50% of RATED THERMAL POWER.~~ 01-18-LS7 01-53-A 01-56-A

(new) Otherwise be in MODE 3 within 12 hours. 01-18-LS7

(new) ACTION 2.1 - With one Channel Inoperable, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied (Note: The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1 and setpoint adjustment): 01-06-LS2 01-17-A

a. Place the inoperable channel in tripped condition within 6 hours, or

b. Be in MODE 3 within 12 hours.

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	~~MODES FOR WHICH SURVEILLANCE IS REQUIRED~~	01-44-A
1. Manual Reactor Trip	N.A.	N.A.	N.A.	~~R(14)~~ R24	N.A.	~~1, 2, 3*, 4*, 5*~~	01-32-LG
2. Power Range, Neutron Flux a. High Setpoint	S	D(~~2, 4~~), ~~M(3, 4)~~, ~~Q(4, 6)~~, R(4, 5, 22)	Q	N.A.	N.A.	~~1, 2~~	01-21-A 01-23-A
b. Low Setpoint	S	R(4, 5, 22)	S/U(1, 20) Q(19, 20)	N.A.	N.A.	~~1###, 2~~	01-22-M 01-23-A
3. Power Range, Neutron Flux High Positive Rate	N.A.	R(4, 5, 22)	Q	N.A.	N.A.	~~1, 2~~	01-39-A 01-23-A
4. Power Range, Neutron Flux High Negative Rate	N.A.	R(4, 5, 22)	Q	N.A.	N.A.	~~1, 2~~	01-39-A 01-23-A
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1, 20) Q(19, 20)	N.A.	N.A.	~~1###, 2~~	01-22-M
6. Source Range, Neutron Flux	S	R(4, ~~5~~)	S/U(1, 8), Q(8, 19)	N.A.	N.A.	~~2##, 3, 4, 5~~	01-26-LG 01-28-A
7. Overtemperature ΔT	S	R (22) M(3, 4) Q(4, 6)	Q	N.A.	N.A.	~~1, 2~~	01-27-LS10 01-23-A
8. Overpower ΔT	S	R (22)	Q	N.A.	N.A.	~~1, 2~~	01-21-A 01-23-A
9. Pressurizer Pressure-Low	S	R (22)	Q	N.A.	N.A.	~~1~~	01-23-A
10. Pressurizer Pressure-High	S	R (22)	Q	N.A.	N.A.	~~1, 2~~	01-23-A
11. Pressurizer Water Level-High	S	R (22)	Q	N.A.	N.A.	~~1~~	01-23-A
12. Reactor Coolant Flow-Low	S	R (22)	Q	N.A.	N.A.	~~1~~	01-23-A

TABLE 4.3-1 (Continued)
REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	~~MODES FOR WHICH SURVEILLANCE IS REQUIRED~~	01-44-A
13. Steam Generator Water Level-Low-Low							
a. Steam Generator Water Level-Low-Low	S	R(22)	Q	N.A.	N.A.	~~1, 2~~	01-23-A
b. RCS Loop ΔT	N.A.	R (22)	Q	N.A.	N.A.	~~1, 2~~	01-23-A
14. DELETED							
15. Undervoltage-Reactor Coolant Pumps	N.A.	R (22)	N.A.	Q(9)	N.A.	~~1~~	01-23-A 01-16-LS40
16. Underfrequency-Reactor Coolant Pumps	N.A.	R (22)	N.A.	Q(9)	N.A.	~~1~~	01-23-A 01-16-LS40
17. Turbine Trip							
a. Low Fluid Oil Pressure	N.A.	R (22)	N.A.	S/U(1, 9)	N.A.	~~1~~	01-36-M 01-23-A
b. Turbine Stop Valve Closure	N.A.	R (22)	N.A.	S/U(1, 9)	N.A.	~~1~~	01-24-LS9
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R24	N.A	~~1, 2~~	
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	R24	N.A.	~~1~~	
20. Reactor Trip System Interlocks							
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	~~2##~~	
~~b. Low Power Reactor Trips Block, P-7~~	~~N.A.~~	~~R(4)~~	R	~~N.A.~~	~~N.A.~~	~~1~~	
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	~~1~~	01-51-LG

TABLE 4.3-[illegible]tinued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	~~MODES FOR WHICH SURVEILLANCE IS REQUIRED~~	01-44-A
20. Reactor Trip System Interlocks (Continued)							
d. Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	~~1~~	
e. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	~~1, 2~~	
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R (22)	R	N.A.	N.A.	~~1~~	01-23-A
21. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, ~~10~~)	N.A.	~~1, 2, 3*, 4*, 5*~~	01-32-LG
(new) Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	N.A.	N.A.	N.A.	M (7)	N.A.		01-14-A
22. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	~~1, 2, 3*, 4*, 5*~~	
23. Seismic Trip	N.A.	R	N.A.	R	R	~~1, 2~~	
24. Reactor Trip Bypass Breaker	N.A	N.A.	N.A.	M(7,15),R~~(16)~~ 24	N.A.	~~1,2,3*,4*,5*~~	01-32-LG

(C)

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

02-01-A

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 ~~and with RESPONSE TIMES as shown in Table 3.3-5.~~ 01-35-LG

APPLICABILITY: As shown in Table 3.3-3.

ACTION: * 01-01-A

a. With an ESFAS Instrumentation Channel or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Values column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value. 02-04-LG

b. With an ESFAS Instrumentation Channel or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value. 02-04-LG

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be ~~demonstrated~~verified OPERABLE by the performance of the Engineered Safety Feature Actuation System Instrumentation Surveillance Requirements specified in Table 4.3-2. 01-03-LS1

24

4.3.2.2 The required ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be ~~demonstrated~~ verified to be within the limit at least once per 18 months on a STAGGERED TEST BASIS**. ~~Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.~~ 01-02-LG 01-03-LS1 02-40-A

(new) * Separate ACTION entry is allowd for each Functional Unit. 01-01-A

(new) ** Not required to be performed for the turbine driven auxiliary feedwater pump until 24 hours after steam generator pressure ≥650 psig. 02-40-A

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	~~TOTAL NO. OF~~ REQUIRED CHANNELS	CHANNELS ~~TO TRIP~~	~~MINIMUM CHANNELS OPERABLE~~	APPLICABLE MODES	ACTION	01-43-A 01-04-LG
3. Containment Isolation (Continued)						
2) Automatic Actuation Logic and Actuation Relays	2	~~1~~	~~2~~	1, 2, 3, 4	14	
3) Containment Pressure-High-High	4	~~2~~	~~3~~	1, 2, 3, 4	17	
c. Containment Ventilation Isolation						02-20-A
1) Automatic Actuation Logic and Actuation Relays	2	~~1~~	~~2~~	1, 2, 3, 4, during CORE ALTERATIONS, during movement of irradiated fuel assemblies within containment.	18, 37	03-14-LS29
2) Deleted						
3) Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.					
4) Containment Ventilation Exhaust Radiation-High (RM-44A and 44B)	2	~~1~~	~~2~~	1, 2, 3, 4.	18, 37	03-14-LS29
4. Steam Line Isolation						
a. Manual	1 manual switch/steam line	~~1 manual switch/steam line~~	~~1 manual switch/ operating steam line~~	1, 2[a], 3[a]~~, 4~~	24	02-07-LS11 02-38-LS35

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	~~TOTAL NO.~~ ~~OF~~ REQUIRED CHANNELS	~~CHANNELS TO TRIP~~	~~MINIMUM CHANNELS OPERABLE~~	APPLICABLE MODES	ACTION	01-43-A 01-04-LG
4. Steam Line Isolation (Continued)						
b. Automatic Actuation Logic and Actuation Relays	2	~~1~~	~~2~~	1, 2[(a)], 3[(a)] .	22	02-07-LS11
c. Containment Pressure-High-High	4	~~2~~	~~3~~	1, 2[(a)], 3[(a)]	17 1	02-15-M 02-07-LS11
d. Steam Line Pressure-Low	3/steam line	~~2/steam line in any steam line~~	~~2/steam line~~	1, 2[(a)], 3#[(a)]	20	02-07-LS11
e. Negative Steam Line Pressure Rate - High		3/steam line	~~2/steam line in any steam line~~	~~2/steam line~~ 3##[(a)]	20	02-07-LS11
5. Turbine Trip & Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	2	~~1~~	~~2~~	1, 2[(b)]	25	02-07-LS11
b. Steam Generator Water Level-High-High	3/stm. gen.	~~2/stm. gen. in any operating stm. gen.~~	~~2/stm. gen. in each oper stm. gen.~~	1, 2[(b)]	~~20~~ 35.2	02-07-LS11 02-08-M

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

01-04-LG
01-43-A

FUNCTIONAL UNIT	~~TOTAL NO.~~ ~~OF~~ REQUIRED CHANNELS	~~CHANNELS TO TRIP~~	~~MINIMUM CHANNELS~~	APPLICABLE MODES ~~OPERABLE~~	ACTION	
6. Auxiliary Feedwater						
a. Manual Initiation	1 manual switch/pump	~~1 manual switch/pump~~	~~1 manual switch/pump~~	1, 2, 3	24	
b. Automatic Actuation Logic and Actuation Relays	2	~~1~~	~~2~~	1, 2, 3	22	
c. Stm. Gen. Water Level-Low-Low						
~~1) Start Motor-Driven Pumps~~						02-09-LG
a. Steam Generator Water Level-Low-Low	3/S.G.	~~2/S.G. in one S.G~~	~~2/S.G. in each S.G.~~	1, 2, 3###	20	|
b. RCS Loop ΔT	4 (1/loop)	~~N.A.~~	~~N.A.~~	1, 2	29	|
~~2) Start Turbine-Driven Pump~~						02-09-LG
add strike-out ~~a. Steam Generator Water Level-Low-Low~~	~~3/S.G.~~	~~2/S.G. in any 2 S.G.~~	~~2/S.G. in each S.G.~~	~~1, 2, 3###~~	~~20~~	|
~~b. RCS loop ΔT~~	~~4 (1/loop)~~	~~N.A.~~	~~N.A.~~	~~1, 2~~	~~29~~	|
d. Undervoltage-RCP Bus ~~Start Turbine-Driven Pump~~	2/bus	~~1/bus on both busses~~	~~1/bus~~	1	35	02-19-LG
e. Safety Injection ~~Start Motor-Driven Pumps~~	See Item 1. above for all Safety Injection initiating functions and requirements.					02-19-LG

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	~~TOTAL NO.~~ ~~OF~~ REQUIRED CHANNELS	~~CHANNELS TO TRIP~~	~~MINIMUM CHANNELS OPERABLE~~	APPLICABLE MODES	ACTION	01-04-LG 01-43-A
7. Loss of Power (4.16 kV Emergency Bus Undervoltage)						02-11-A
a. First Level				1, 2, 3, 4**		02-36-M
1) Diesel Start	1/Bus	~~1/Bus~~	~~1/Bus~~		16	
2) Initiation of Load Shed	2/Bus	~~2/Bus~~	~~2/Bus~~		~~16~~15	02-48-LS2
b. Second Level				1, 2, 3, 4**		02-36-M
1) Undervoltage Relays	2/Bus	~~2/Bus~~	~~2/Bus~~		~~16~~15	02-48-LS2
2) Timers to Start Diesel	1/Bus	~~1/Bus~~	~~1/Bus~~		16	
3) Timers to Shed Load	1/Bus	~~1/Bus~~	~~1/Bus~~		16	
8. Engineered Safety Features Actuation System Interlocks						
a. Pressurizer Pressure, P-11	~~3~~2	~~2~~	~~2~~	1, 2, 3	21	01-37-A
b. DELETED						
c. Reactor Trip, P-4	2	~~2~~	~~2~~	1, 2, 3	23	
9. Residual Heat Removal pump trip on ~~low~~ RWST level-low	3	~~2~~	~~2~~	1, 2, 3, 4	~~20.1~~ 36	02-29-M

(8)

ACTION 19 - With ~~the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement~~ one channel inoperable, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. 01-04-LG

ACTION 20 - With ~~the number of OPERABLE channels one less than the Total Number of Channels~~ one channel inoperable, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: 02-08-M 01-43-A

a. The inoperable channel is placed in the tripped condition within 6 hours, ~~and~~ or

b. Be in MODE 3 in 12 hours and in MODE 4 in 18 hours. NOTE: The ~~Minimum Channels OPERABLE requirement is met; however, the~~ inoperable channel or one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1. 01-04-LG

ACTION 20.1 - With one channel inoperable, STARTUP and/or POWER OPERATION may proceed (new) provided the following conditions are satisfied: 02-29-M

a. The inoperable channel is placed in the bypassed condition within 6 hours, and the inoperable channel is returned to an OPERABLE status within 72 hours, or

b. Immediately enter 3.0.3.

NOTE: One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

ACTION 20.2 - With one channel inoperable, STARTUP and/or POWER OPERATION may (new) proceed provided the following conditions are satisfied: 02-08-M

a. The inoperable channel is placed in the tripped condition within 6 hours, or

b. Be in MODE 3 in 12 hours and in MODE 5 in 42 hours. NOTE: The inoperable channel or one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

ACTION 36 - With the number of OPERABLE channels one less than the Total Number of Channels, within 6 hours place the inoperable channel in cut-out and restore the inoperable channel to OPERABLE status within 72 hours; or be in at least Hot Standby within the next 6 hours and be in Cold Shutdown within the next 30 hours. 02-29-M

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 21 - With the number of OPERABLE channels less than the ~~Minimum Number of~~ Required Channels OPERABLE, within 1 hour determine ~~by observation of the associated permissive annunciator window(s)~~ that the interlock is in its required state for the existing plant condition, or ~~apply Specification 3.0.3~~ be in MODE 3 in 7 hours and MODE 4 in 13 hours. 01-04-LG 02-14-M 01-52-LG

ACTION 22 - With the number of OPERABLE Channels one less than the ~~Minimum~~ Required Channels ~~OPERABLE requirement~~, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE. 01-04-LG

ACTION 23 - With the number of OPERABLE channels one less than the ~~Total Number of~~ Required Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours. 01-43-A

ACTION 24 - With the number of OPERABLE channels one less than the ~~Total Number of~~ Required Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated pump or valve inoperable and take the ACTION required by Specification 3.7.1.5 or 3.7.1.2 as applicable. 01-43-A

ACTION 25 - With the number of OPERABLE channels one less than the ~~Minimum~~ Required Channels ~~OPERABLE requirement~~, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE. 01-04-LG

ACTION 29 - With the number of OPERABLE channels less than the ~~Total Number of~~ Required Channels, STARTUP and/or POWER OPERATION may proceed provided that within 6 hours, for the affected RCS Loop Delta-T channel(s), either: 01-43-A

a. The Trip Time Delay threshold power level for zero seconds time delay is adjusted to 0% RTP, or

b. With the number of OPERABLE channels one less than the ~~Total Number of~~ Required Channels, the affected Steam Generator Water Level-Low-Low channels are placed in the tripped condition. 01-43-A

delete BOLD & strike-out

ACTION 35 - With the number of OPERABLE channels one less than the ~~Total Number of~~ Required Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: 01-43-A

a. The inoperable channel is placed in the trip condition within 6 hours, ~~and~~ or be in MODE 2 in 12 hours. 02-08-M

b. ~~The Minimum Channels OPERABLE requirement is met; however~~. The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1. 01-04-LG

ACTION 35.2 - With the number of OPERABLE channels one less than the Required Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: 02-08-M

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

word should be coincident

02-28-LG

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
2. Containment Spray ~~(Containment with SI Signal)~~		
a. Manual Initiation	N.A.	N.A
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A
c. Containment Pressure-High-High	≤ 22 psig	≤ 22.3 psig
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual	N.A.	N.A
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
b. Phase "B" Isolation		
1) Manual	N.A.	N.A
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A
3) Containment Pressure-High-High	≤ 22 psig	≤ 22.3 psig

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Loss of Power (4.16 kV Emergency Bus Undervoltage)		02-11-A
a. First Level		
1) Diesel Start	$\leq$ 0.8 second time delay and $\geq$ 2583 volts with a $\leq$ 10 second time delay One relay	$\geq$ 0 volts with a $\geq$ 0 volts with a $\leq$ 0.8 second time delay and $\geq$ 2583 volts with $\leq$ 10 second time delay One relay
2) Initiation of Load Shed	$\geq$ 0 volts with a $\leq$ 4 second time delay and $\geq$ 2583 volts with a $\leq$ 25 second time delay with one relay $\geq$ 2870 volts, instantaneous	$\geq$ 0 volts with a $\leq$ 4 second time delay and $\geq$ 2583 volts with a $\leq$ 25 second time delay with one relay $\geq$ 2870 volts, instantaneous
b. Second Level		
1) Diesel Start	$\leq$ 10 second time delay	$\geq$ 3785 volts with a $\geq$ 3785 volts with a $\leq$ 10 second time delay
2) Initiation of Load Shed	$\geq$ 3785 volts with a $\leq$ 20 second time delay	$\geq$ 3785 volts with a $\leq$ 20 second time delay
8. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure, P-11	$\leq$ 1915 psig	$\leq$ 1920.6 psig
b. DELETED		
c. Reactor Trip, P-4	N.A.	N.A.
9. Residual Heat Removal pump trip on ~~low~~ RWST level – low	$\leq$ ~~33% level~~ 32.8	$\leq$ ~~32.9% level~~ 33.68 and $\geq$ 31.44%

02-29-M

NOTE 1: Time constants utilized in the lead-lag controller for Steam Pressure - Low are τ_1 = 50 seconds and τ_2 = 5 seconds.

NOTE 2: Steam Generator Water Level Low-Low Trip Time Delay

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

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

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

B1 = -0.007128
B2 = +0.8099
B3 = -31.40
B4 = +464.1

NOTE 3: Time constants utilized in the rate-lag controller for Negative Steam Line Pressure Rate-High are τ_3 = 50 seconds and τ_4 = 50 seconds.

TABLE [illegible]-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

01-44-A

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	~~MODES FOR WHICH SURVEILLANCE IS REQUIRED~~	
1. Safety Injection, ~~(Reactor Trip Feedwater Isolation, Start Diesel Generators, Containment Fan Cooler Units, and Component Cooling Water)~~									02-19-LG
a. Manual Initiation	N.A.	N.A.	N.A.	R24	N.A.	N.A.	N.A.	~~1, 2, 3, 4~~	
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	~~1, 2, 3, 4~~	
4 c. Containment Pressure-High	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	~~1, 2, 3,~~	01-23-A
d. Pressurizer Pressure-Low	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	~~1, 2, 3~~	
e. DELETED									01-23-A
f. Steam Line Pressure-Low	S *	R(6)	Q	N.A.	N.A.	N.A.	N.A.	~~1, 2, 3,~~	01-23-A
2. Containment Spray ~~(coincident with SI signal)~~									02-28-LG
4 a. Manual Initiation	N.A.	N.A.	N.A.	R24	N.A.	N.A.	N.A.	~~1, 2, 3,~~	
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	~~1, 2, 3, 4~~	
c. Containment Pressure-~~High-~~High ~~1, 2, 3, 4**~~	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.		01-23-A

~~* These changes from License Amendments 84 & 83.~~

~~** These changes from License Amendments 89 & 88.~~

C
E

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

01-44-A

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALI-BRATION	CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERA-TIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	~~MODES FOR WHICH SURVEILLANCE IS REQUIRED~~
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual	N.A.	N.A.	N.A.	R24	N.A.	N.A.	N.A.	~~1, 2, 3, 4~~
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	~~1, 2, 3, 4~~
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual	N.A.	N.A.	N.A.	R24	N.A.	N.A.	N.A.	~~1, 2, 3, 4~~
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	~~1, 2, 3, 4~~
3) Containment Pressure-High-High	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	~~1, 2, 3, 4~~

01-23-A

02-20-A

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALI-BRATION	CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERA-TIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	~~MODES FOR WHICH SURVEILLANCE IS REQUIRED~~
c. Containment Ventilation Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	~~1, 2, 3, 4~~
2) Deleted								
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4) Containment Ventilation Exhaust Radiation-High (RM-44A and 44B)	S	R(6)	Q(2)	N.A.	N.A.	N.A.	N.A.	~~1, 2, 3, 4~~

01-23-A

02-35-A

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

01-44-A

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR ~~WHICH SURVEILLANCE IS REQUIRED~~	
4. Steam Line Isolation									
a. Manual	N.A.	N.A.	N.A.	R24	N.A.	N.A.	N.A.	~~1, 2, 3~~	
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	~~1, 2, 3~~	
3 c. Containment Pressure-High-High	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	~~1, 2,~~	01-23-A
d. Steam Line Pressure-Low	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	~~1, 2, 3~~	01-23-A
e. Negative Steam Line Pressure Rate-High	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	~~3(3)~~	01-23-A
5. Turbine Trip and Feedwater Isolation									
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	~~1, 2~~	
b. Steam Generator Water Level-High-High	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	~~1, 2~~	01-23-A
6. Auxiliary Feedwater									
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	~~1, 2, 3~~	
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	R	~~1, 2, 3~~	
c. Steam Generator Water Level-Low-Low									
1) Steam Generator Water Level-Low-Low	S	R(6)	Q	N.A.	N.A.	N.A.	N.A.	~~1, 2, 3(5)~~	01-23-A
2) RCS Loop ΔT	N.A.	R(6)	Q	N.A	N.A	N.A	N.A	~~1, 2~~	01-23-A

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	~~MODES FOR WHICH SURVEILLANCE IS REQUIRED~~	01-44-A
6. Auxiliary Feedwater (Continued)									
d. Undervoltage - RCP	N.A.	R(6)	N.A.	R	N.A.	N.A.	N.A.	~~1~~	01-23-A
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.								
7. Loss of Power									02-11-A
a. 4.16 kV Emergency Bus Level 1	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	~~1, 2, 3, 4~~	
b. 4.16 kV Emergency Bus Level 2	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	~~1, 2, 3, 4~~	
8. Engineered Safety Feature Actuation System Interlocks									
a. Pressurizer Pressure, P-11	N.A.	R(6)	Q	N.A.	N.A.	N.A.	N.A.	~~1, 2, 3~~	01-23-A
b. DELETED									
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R 24 (6)	N.A.	N.A.	N.A.	~~1, 2, 3~~	01-23-A
9. Residual Heat Removal pump trip ~~on low~~ RWST level-low	S	~~R(6)~~ R24	N.A.	R24	N.A.	N.A.	N.A.	~~1, 2, 3, 4~~	01-29-M 01-23-A

TABLE NOTATIONS

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) For the Containment Ventilation Exhaust Radiation - High monitor only, a CHANNEL FUNCTIONAL TEST shall be performed at least once every ~~31~~92 days. 02-35-A
(3) Trip function automatically blocked above P-11 (Pressurizer Pressure Interlock) setpoint and is automatically blocked below P-11 when Safety Injection on Steam Line Pressure-Low is not blocked.
(4) DELETED
(5) For Mode 3, the Trip Time Delay associated with the Steam Generator Water Level-Low-Low channel must be less than or equal to

T/ 3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

	CHANNEL CHECK	CHANNEL CALIBRATION	ACTUATION LOGIC TEST	TADOT	CHANNEL FUNCTIONAL TEST	MASTER RELAY TEST	SLAVE RELAY TEST	~~MODES FOR WHICH SURVEILLANCE IS REQUIRED~~	01-44-A
1. Fuel Handling Building									03-01-A
(New) Manual	NA	NA	NA	R(c)	NA	NA	NA		03-08-M
a. Storage Area									
1) Spent Fuel Pool	S	R	NA	NA	Q	NA	NA	~~*~~	
2) New Fuel Storage	S	R	NA	NA	Q	NA	NA	~~*~~	
b. Gaseous Activity									
Fuel Handling Building Ventilation Mode Change(a)	S	R	NA	NA	Q	NA	NA	~~*~~	
2. Control Room									03-01-A
Ventilation Mode Change	S	R			Q			All	
a. Manual Initiation	NA	NA	NA	R(c)	NA	NA	NA		03-08-M
b. Automatic Actuation Logic and Actuation Relays	NA	NA	M(b)	A	NA	M(b)	R		
c. Control Room Atmosphere Air Intake Radiation	S	R	NA	NA	Q	NA	NA		
3. Containment									
a. Gaseous Activity									
1) Deleted									
2) RCS Leakage	S	R	NA	NA	Q	NA	NA	~~1, 2, 3, 4~~	03-01-A
3) Containment Ventilation Isolation	S	R	NA	NA	Q	NA	NA	6	
(RM-44A or 44B)	S	R			Q				
b. Particulate Activity									
1) Containment Ventilation Isolation (RM-44A or 44B)	S	R	NA	NA	Q	NA	NA	6	
2) RCS Leakage	S	R	NA	NA	Q	NA	NA	~~1, 2, 3, 4~~	03-01-A

Should be Red Lined

~~* With fuel in the spent fuel pool or new fuel storage vault.~~ 03-03-LG

(a) The requirements for Fuel Handling Building Ventilation Mode Change are applicable following installation of RM-45A and 45B. 03-01-A

(New) (b) Each train shall be tested at least once every 62 days on a STAGGERED TEST BASIS. 03-08-M

(New) (c) Verification of setpoint is not required. 03-08-M

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C & E

DESCRIPTION OF CHANGES TO TS SECTION 3/4.3

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

CHANGE NUMBER	NSHC	DESCRIPTION
01-01	A	A Note, "Separate Condition entry is allowed for each Function," is added to the ACTIONS for the Reactor Trip System, ESFAS, [Remote Shutdown also applies to each required ASP control], and Accident Monitoring Instrumentation. This change clarifies those situations where the current TS ACTION Statements are not uniquely associated with a particular Function or where the required channels are specified on a per steam line, per loop, per SG, per bus, etc., basis. This change is consistent with current operating practices and NUREG-1431. [ ]
01-02	LG	The CTS require that response time testing be performed on each reactor trip and ESFAS function every ~~18~~ [24] months and that alternate trains be tested in successive tests. The CTS description of the channel testing protocol matches the improved TS definition of STAGGERED TEST BASIS. However, several trip functions do not require response time testing, as indicated by N.A. in the tables of response time limits [(presently located in Tables 3.3-2 and 3.3-5 of the CTS, which are being to the FSAR per CN 01-35-LG)]. The improved TS specify that required response time testing be performed on a STAGGERED TEST BASIS and do not impose any requirements as to which train should be tested. Therefore, the word "requirement" is added to the CTS and the requirement to ensure that each train is tested every [48] ~~36~~ months is moved to the Bases for ITS SR 3.3.1.16 and SR 3.3.2.10.
01-03	LS1	In CTS SR 4.3.1.2 and 4.3.2.2, the active verb is changed from "demonstrated" to "verified." This allows Reactor Trip System and ESFAS sensor response time verifications to be performed per WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements." This change is consistent with Traveler TSTF-111 Rev. 1, which revises the Bases for ITS SR 3.3.1.16 and SR 3.3.2.10 to allow the elimination of pressure sensor response time testing.

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CHANGE NUMBER	NSHC	DESCRIPTION
01-21	A	The monthly and quarterly channel calibrations associated with Notes (3), (4), and (6) of CTS Table 4.3-1 have been moved from the Power Range Neutron Flux-High Setpoint Function to the Overtemperature [ΔT] Function. This change clarifies the relationship of these surveillances to the f_1 (ΔI) penalty portion of the Overtemperature [ΔT] Function. The primary purpose of these surveillances is to verify correct f_1 (ΔI) input to Overtemperature [ΔT]. Although these surveillances affect all power range neutron flux channels, and appropriate action must be taken for any affected power range neutron flux channel, this change groups the surveillances with the most appropriate reactor trip function for OPERABILITY concerns. [ ] The applicable portions of CTS Table 4.3-1 Notes (3) and (6) are incorporated directly into ITS SR 3.3.1.3 and SR 3.3.1.6, as discussed in CN 1-25-A. Note (4) has been deleted from the daily, monthly, and quarterly surveillances associated with Notes (2), (3), and (6) of CTS Table 4.3-1 since these surveillances are not CHANNEL CALIBRATIONS, rather they are comparisons and adjustments as needed. These changes are consistent with NUREG-1431.
01-22	M	Quarterly COTs have been added to CTS Table 4.3-1 for the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux trip functions in the event extended operation within their APPLICABILITY (i.e., MODE 1 below P-10 and MODE 2) takes place. The CTS only require a COT prior to startup for these functions. New Note [(19)] has been added to require that the new quarterly COT be performed within 12 hours after reducing power below P-10 for the power range and intermediate range instrumentation (P-10 is the dividing point marking the APPLICABILITY for these trip functions), if not performed within the previous 92 days. [In addition,new Note (20) has been added] such that the P-6 and P-10 interlocks are verified to be in their required state during all COTs on the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux trip functions. These changes are consistent with NUREG-1431 and traveler WOG-106.
01-23	A	This change adds new Note [(22)] to CTS Table 4.3-1 and new Note [(56)] to CTS Table 4.3-2 that explicitly require the 18-month calibrations to include verifications of affected time constants, consistent with NUREG-1431.

E

CHANGE NUMBER	NSHC	DESCRIPTION
01-43	A	The "Total Number of Channels" columns in CTS Tables 3.3-1 and [3.3-3] and the ["Minimum Channels OPERABLE"] column in CTS Table [3.3-6] and the reference to them in the ACTIONS are relabeled as the "Required Channels" consistent with NUREG-1431. ACTION Statements are revised to use the ITS terminology, "Required Channels". Changing the column titles is purely administrative. The numbers in the columns are adjusted, if necessary. Where the numbers are adjusted, those changes are described in different CNs.
01-44	A	and 4.3-3 The "MODES For Which Surveillance Is Required" columns in CTS Tables 4.3-1 and 4.3-2 [/] are deleted since this information is enveloped by CTS Tables 3.3-1 and [3.3-3] and 3.3-6 and is redundant given the integrated OPERABILITY/ SR format in improved TS Tables 3.3.1-1 and 3.3.2-1.
01-45	M	The Overtemperature [ΔT], Overpower [ΔT], Pressurizer Pressure - High, and Steam Generator Water Level - Low-Low trip functions, which currently reference ACTION Statement [6], are now referenced to new ACTION Statement [2.1], consistent with ITS 3.3.1 Condition E. This change is more restrictive since one less hour is available under new ACTION Statement [2.1] than under the combination of current ACTION Statement [6] and LCO 3.0.3.
01-46	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-47	A	A new note (f) is added and applied to the Functional Unit 6.c. The note is for clarification only as the CTS Table 3.3-1 indicates that only 1 channel is required to be OPERABLE and that there is no trip function under these conditions.
01-48	LS4	CTS ACTION 7 is revised to allow a reduction in power below P-9 in lieu of tripping the inoperable channel. The ACTION is also revised to allow bypassing a tripped channel for four hours for surveillance testing other channels. Note (j) is added to Table 3.3-1, Applicable Modes for Functional Unit 17.a and b that states that the requirements are only applicable above P-9.
01-49	LS18	CTS ACTION 9 is deleted and revised ACTION 6 is used which allows a power reduction below P-7 in lieu of tripping the inoperable channel. Note (g) is added that specifies that Functional Unit 19 of the CTS does not have to be applied until the power level associated with P-7 is reached. ACTION 6 also allows the tripped channel to be bypassed for up to 4 hours to perform surveillance testing on other channels.

E

CHANGE NUMBER	NSHC	DESCRIPTION
01-50	A	ACTION [28] of the CTS duplicates CTS ACTION [6] and is deleted.
01-51	LG	This change moves the description of the P-7 inputs, i.e., P-10 and P-13, to the Bases since they are duplicated by Functional Units [20.e and 20.f], The Required Channels column for P-7 lists "1 per train" since this is a more appropriate convention for a logic function. These changes are consistent with NUREG-1431. [This change also deletes the surveillance requirements for P-7 per CN 3.3-54 in the ITS since the COTs and channel calibration apply to P-10 and P-13 not to the P-7 logic function.]
01-52	LG	This change moves the specifics on how to verify permissive functions of ACTIONS [8] and [21] to the Bases, consistent with NUREG-1431. This information is more appropriately controlled outside of the TS while the underlying requirement to verify proper permissive operation is unchanged.
01-53	A	CTS Table 3.3-1 ACTION Statement [2.c] is revised to be consistent with ITS SR 3.2.4.2, as discussed in CN 4-04-LS-12 in the 3/4.2 package.
01-54	LS37	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
01-55	LS39	APPLICABILITY Note [*] and ACTION Statement [11] for Functional Units [1, 6.b, 20, and 21] of CTS Table 3.3-1 are modified to provide an alternative to opening the reactor trip breakers (RTBs) while still assuring that the function and intent of opening the RTBs is met. As currently worded, these ACTION Statements result in a feedwater isolation signal (FWIS) when in MODE 3 with a T_{avg} less than [554°F. FSAR Table 7.3-3 and FSAR Figure 7.2-1 (sht. 13) detail the FWIS generation on the coincidence of P-4 and low T_{avg}.] A more generic action, which assures the rods are fully inserted and cannot be withdrawn, replaces the specific method of precluding rod withdrawal. The revised APPLICABILITY and ACTION Statements still assure rod withdrawal is precluded. This change does not involve any safety impact and is consistent with traveler TSTF-135.
01-56	A	The DCPP CTS 3.3.1 ACTION 2.c requires that power be reduced to less than 75% or that SR 4.2.4.2 be performed whenever power is ≥ 50%. This power level requirement should be ≥ 75% since if power is decreased below 75% per the first part of Action 2.c, the required ACTION is complete and in addition, SR 4.2.4.2 is only required for power levels ≥ 75% with one power range detector inoperable.

CHANGE NUMBER	NSHC	DESCRIPTION
02-07	LS11	[Note (a) is added to CTS Table 3.3-3 for the Steam Line Isolation Functional Units 4.a, 4.b, 4.c, 4.d, and 4.e to state that the LCO requirements are not applicable in MODES 2 and 3 when the MSIVs are closed and deactivated]. Note [(b)] is added to CTS Table [3.3-3] for the Feedwater Isolation and Turbine Trip Function [Functional Units 5.a, and 5.b] to state that the LCO requirements are not applicable when the [MFIVs, MFRVs or the associated bypass valves] are closed [and deactivated or isolated by a closed manual valve]. When these valves are closed [and deactivated or isolated by a closed manual valve], they are already performing their safety function. These changes are consistent with NUREG-1431.
02-08	M	[This change revises ACTION 20 and 35 in CTS Table 3.3-3 and adds new ACTION 20.2 and 35.2 which are applicable to Functional Units 1.c, 1.d, 1.f, 4.c 4.d, 4.e, 5.b, 6.c.1)a, and 6.d]. These ACTION Statements, written to reflect the APPLICABILITY of the affected channels and consistency with ITS 3.3.2 [Conditions D and I], are more restrictive, by one hour, than the current ACTION Statement[s] which invoke[ ] LCO 3.0.3 if the inoperable channel is not placed in trip within 6 hours.
02-09	LG	Separate ESFAS entries for the motor-driven and turbine-driven auxiliary feedwater pumps are no longer necessary, consistent with NUREG-1431. The only difference in the requirements (an SR 4.0.4 exception for response time testing of the turbine-driven auxiliary feedwater pump) has been addressed in the ITS by a Note in Surveillance Requirement 3.3.2.10. [The details of which actuation signal starts which pump is moved to the Bases for SI and RCP undervoltage].
02-10	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-11	A	The Functional Unit for Loss of Power [CTS 7.a, 7.b] is moved to improved TS 3.3.5.
02-12	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-13	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-14	M	This change modifies ACTION Statement [21] for permissive P-11 [ ] to provide specific shutdown requirements to exit APPLICABILITY in lieu of applying LCO 3.0.3. This change is more restrictive by one hour, consistent with NUREG-1431.

(B) & (E)

CHANGE NUMBER	NSHC	DESCRIPTION
02-28	LG	This change moves information inserted by LA 114/112 on containment spray and safety injection coincidence to the Bases, consistent with NUREG-1431.
02-29	M	A new functional unit 9 is added, per ~~a~~ License Amendment Request 97-10 that incorporates ACTION ~~20.1~~ 36 (new) and Surveillances for the residual heat removal (RHR) pump trip from low refueling water storage tank level.
02-30	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-31	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-32	LS23	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-33	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-34	LS34	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-35	A	Note (2) of the CTS Table 4.3.2 is revised. The testing frequency was relaxed from monthly (M) to quarterly (Q) via License Amendment 102/101, but the Note was not revised nor was it shown as applicable to Functional Unit 3.c.4).
02-36	M	This change revises the APPLICABILITY of Functional Unit 7 to require OPERABILITY when the associated DG is required to be OPERABLE by LCO 3.8.2 of the ITS. This change is consistent with NUREG-1431.
02-37	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-38	LS35	Delete MODE 4 APPLICABILITY from the Manual Initiation of MSIVs since the valves are not required to be OPERABLE in MODE 4 per CTS 3.7.1.5 or ITS 3.7.2.
02-39	LG	Move valve numbers in CTS ACTION 18 dealing with containment ventilation isolation to the Bases, consistent with NUREG-1431.
02-40	A	This administrative change affects the manner in which the CTS 4.0.4 exception for testing the TDAFW pump is presented. The exception allows entry into MODE 3 to perform the TDAFW pump response time testing. In NUREG-1431, the CTS 4.0.4 exception from [CTS 3.7.1.2 has been interpreted so that it allows response time testing to be deferred as] is reflected in the ITS SR 3.3.2.10 NOTE.

(E)

CHANGE NUMBER	NSHC	DESCRIPTION
02-41	LS36	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-42	LS38	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-43	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-44	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-45	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-46	LS42	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-47	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-48	LS28	A new Action 15 is added and applied to Functional Unit 7. a. 2) and 7. b. 1) that allows the affected Emergency Diesel Generator to be declared inoperable and requires entry into Specification 3.8.1.1 when more than one relay per bus is inoperable. Current ACTION 16 does not address the above situation and requires entry into LCO 3.0.3. This change is consistent with NUREG-1431.
03-01	A	The requirements of this specification [CTS 3.3.3.1] are moved to [four] separate specifications in the improved TS. The RCS Leakage Detection requirements are moved to improved TS 3.4.15. [The Fuel Building requirements are moved to improved TS 3.3.8.] The Control Room requirements are moved to improved TS 3.3.7. [The Containment Ventilation Isolation requirements are moved to improved TS 3.3.6.]
03-02	M	The requirements stipulated in ACTION [a] are moved to ITS Tables [3.3.6-1, 3.3.7-1 and 3.3.8-1], with explicit direction contained in the ITS ACTIONS Bases. The 4 hour AOT for setpoint adjustment is eliminated.
03-03	LG	The requirements associated with the criticality monitors are moved to a licensee controlled document. These monitors are required by 10CFR70.24; however, there is no requirement for [them] to be in the Technical Specifications [as criticality monitors. They are retained, however, as initiators of the Iodine Removal mode of the FHBVS for a fuel handling accident until RM-~~44A~~ 45A and ~~44B~~ 45B are installed in accordance with License Amendment 70/69]. Since Part 70 is invoked in the operating licens~~d~~e, these monitors will be retained in the plant design.

E

CHANGE NUMBER	NSHC	DESCRIPTION
03-04	M	This change adds the APPLICABILITY for movement of irradiated fuel assemblies consistent with NUREG-1431. The CTS APPLICABILITY of "All" MODES does not cover the movement of irradiated fuel assemblies when the core is offloaded.
03-05	LS14	ACTION Statement [34] for the [Control Room Air Intake] [ ] radiation monitors have extended Completion Times, from [1 hour] to 7 days for one required channel inoperable, consistent with NUREG-1431.
03-06	A	ACTION [c] of CTS LCO 3.3.3.1 is revised to state the Specification 3.0.3 exception is [retained only for the Fuel Handling Building Radioactivity Instrumentation]. The LCO 3.0.3 exception is not needed in ITS 3.3.7 or ITS 3.4.15 since Required Actions are provided with the appropriate remedial measures for all combinations of failures, including shutdown actions, or reference is made to the associated plant system TS for the systems affected by the inoperability of the radiation monitors. [ ].
03-07	LS16	The APPLICABILITY for the Fuel [Handling] Building Exhaust radiation monitors has been revised to read "during movement of irradiated fuel assemblies in the fuel [handling] building." [The REQUIRED CHANNELS for Instrument 1.b. has been revised from one as specifed by the CTS to two as specifed by NUREG-1431 to provide protection against a single failure that could prevent the transfer of the FHBVS to the iodine removal mode.]
03-08	M	The CTS have been revised to include manual initiation of the fuel handling building and manual and automatic initiation of the control room pressurization system. These systems are not classified as ESF functions in the CTS even though CTS surveillance 4.7.5.1e.2) requires that the CRVS automatically switches to the pressurization mode on a Phase "A" signal. The FHBVS is not an ESF function since its only function is to mitigate a fuel handling accident. This revision incorporates the Actuation Logic, Master Relay, and Slave Relay Tests included in NUREG-1431 for the CRVS and the TADOT for the manual actuation of both systems. The automatic actuation tests are conducted as part of the CTS, and the relay tests are currently performed even though not specifically called out in the CTS.
03-09	LS-24	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-10	LG	The DCPP descriptive information related to the Required Channels per normal intake is moved to the Bases.

(E)

CHANGE NUMBER	NSHC	DESCRIPTION
03-11	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-12	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-13	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-14	LS29	This proposed change adds an ACTION and an allowed outage time of 4 hours for one inoperable Containment Ventilation Radiation instrumentation or actuation channel. The CTS via ACTIONS 18 and 33 requires that for one or two instruments or channels inoperable that CTS 3.6.3 or 3.9.9 be entered. The revised TS will require that ITS 3.6.3 or 3.9.4 be entered if the instrument or channel cannot be returned to an OPERABLE status within the revised AOT. This change is consistent with the requirements of NUREG-1431.
03-15	M	This change revises CTS ACTION 34 to require appropriate MODE changes or condition changes for the CRVS with one inoperable normal intake monitor and new ACTION 36 specifies actions for two inoperable normal intake monitors. The CTS requires that if the required ACTIONs for one inoperable CRVS monitor is not met that LCO 3.0.3 be entered. In addition, the CTS does not specify a required action if both monitors are inoperable. NUREG-1431 requires that for the above conditions that appropriate actions be taken to place the plant in acondition of non-APPLICABILITY. These ACTIONs specify a shutdown requirement for MODES 1-4 that is one hour less than LCO 3.0.3, and immediate ACTION for inoperability in MODE 5 or 6, and immediate action for inoperability during fuel movement. These changes are consistent with NUREG-1431. Refer also to CN 03-08-M, CN 03-04-M, and CN 3.3-51.
03-16	A e	NOT USED ITS 3.3.6 for DCPP includes MODES 1-4 and during movement of irradiated fuel assemblies within containment, in addition to MODE 6, in the LCO APPLICABILITY. These requirements are inferred in CTS 3.6.3 and are repeated here for clarity. e
03-17	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
04-01	R	DCPP LCO 3.3.3.2, Movable Incore Detectors, is relocated to a licensee controlled document, see Attachment 21, page 11.
05-01	R	DCPP LCO 3.3.3.3, Seismic Instrumentation, is relocated to a licensee controlled document, see LAR 95-07.

(E)

CHANGE NUMBER	NSHC	DESCRIPTION
08-03	A	This change revises CTS Table [3.3-10] to clarify the number of channels required to be Operable. This is an administrative change which deletes the "Minimum Channels Operable" column [ ]. The required ACTIONs are now based on one channel inoperable or two channels inoperable, rather than "less than the Total Number" or "less than Minimum Number." This change is consistent with NUREG-1431.
08-04	LS17	Consistent with NUREG-1431 (ITS 3.3.3 Required ACTIONs C.1, E.1, and G.1), this change deletes the requirement to initiate an alternate means of monitoring within 72 hours when two channels of Containment Radiation Level ~~[or RVLIS]~~ are inoperable as specified in CTS [3.3.3.6 ACTION d. In addition, a special report is required within 14 days that identifies the alternate method of monitoring the appropriate parameter(s), as well as the current special report requirements].
08-05	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-06	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-07	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-08	LS27	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-09		Not Used.
08-10		Not Used.
08-11	LS30	This change revises the DCPP CTS 3.3.3.6 to conform to NUREG-1431 and revises CTS Table 3.3-10 to both add and delete instruments per the Reviewer's Note on ISTS Table 3.3.3-1.
09-01	LG	The explosive gas monitoring instrumentation will be controlled by the Explosive Gas Monitoring Program established in accordance with ITS 5.5.12, see Attachment 21, page 15.
10-01	R	The Turbine Overspeed Protection System is relocated to a licensee controlled document, see LAR 95-07.
11-01	R	LCO 3.3.3.7, Chlorine Detection Systems, is relocated to a licensee controlled document, see LAR 95-07.

[illegible]

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-01 A	A Note, "Separate Condition entry is allowed for each Function," is added to the ACTIONS for the Reactor Trip System, ESFAS, [ ] and Accident Monitoring Instrumentation. This change clarifies those situations where the CTS ACTION Statements are not uniquely associated with a particular Function or where the required channels are specified on a per steam line, per loop, per SG, per bus, etc., basis.	Yes	Yes	Yes, see also CN 2-46-LS-42.	Yes, see also CN 2-46-LS-42.
01-02 LG	The improved TS specify that required response time testing be performed on a STAGGERED TEST BASIS and do not impose any requirements as to which train should be tested. The requirement to ensure that each train is tested every ~~36~~ 48 months is moved to the Bases for SR 3.3.1.16 and SR 3.3.2.10.	Yes	Yes	Yes	Yes
01-03 LS1	Changing "demonstrated" to "verified" allows Reactor Trip System and ESFAS sensor response time verifications to be performed per WCAP-13632-P-A Revision 2. This change is consistent with traveler TSTF-111.	Yes	No, see CN 1-58-A.	Yes	Yes
01-04 LG	In CTS Tables [3.3-1 and 3.3-3], the ["Channels to Trip" and] "Minimum Channels OPERABLE" columns are deleted. [ ] The ACTION Statements have been revised accordingly.	Yes	Yes	Yes	Yes
01-05 A	The LCO 3.0.4 exception [footnote #] in CTS Table 3.3-1 is deleted entirely. ACTION Statement [8] in CTS Table 3.3-1 permits continued operation for an unlimited period of time. Therefore, no exception to ITS LCO 3.0.4 is needed for this ACTION Statement. []	Yes	No, not in CTS.	Yes	Yes

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-42 M	This change adds direction to ACTIONS 7, 11, and 13 in CTS Table 3.3-1 to be in MODE 3 within 12 hours, in lieu of LCO 3.0.3 entry, if inoperable EAM/TTD timer(s) or channel(s) aren't tripped within 6 hours.	No, not in current design or TS.	No, not in current design or TS.	No, not in current design or TS.	Yes
01-43 A	The "Total Number of Channels" columns in CTS Tables 3.3-1 and [3.3-3] and the ["Minimum Channels OPERABLE"] column in CTS Table [3.3-6] and the references to them in the ACTIONS are relabeled as the "Required Channels" consistent with NUREG-1431 Rev. 1. ACTION Statements have been revised accordingly.	Yes	Yes	Yes	Yes
01-44 A	This change deletes the "MODES For Which Surveillance Is Required" column in CTS Tables 4.3-1, 4.3-2[and 4.3-3]. e And	Yes	Yes	Yes	Yes
01-45 M	The Overtemperature [ΔT], Overpower [ΔT], Pressurizer Pressure - High, and Steam Generator Water Level - Low-Low trip functions, which currently reference ACTION Statement [6], are now referenced to new ACTION Statement [2.1], consistent with ITS 3.3.1 Condition E. This change is more restrictive since one less hour is available under new ACTION Statement [2.1] than under the combination of current ACTION Statement [6] and LCO 3.0.3.	Yes	Yes	Yes	Yes
01-46 A	ACTION Statement 13 of CTS Table 3.3-1 and ACTION Statement 36 of CTS Table 3.3-3 are revised to reflect operating and testing options that have existed since the SG Water Level Low-Low EAM/TTD design was implemented, but were not listed in the TS since they were not necessarily the options of choice.	No, not in current design or TS.	No, not in current design or TS.	No, not in current design or TS	Yes, reviewed in OL Amendment No. 43 dated April 14, 1989.

[illegible]

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-04 LG	The requirements stipulated in ACTIONS a and b are moved to ITS Table 3.3.2-1, with explicit direction contained in the ITS ACTIONS Bases.	Yes	Yes	Yes	Yes
02-05 M	The Functional Unit for Containment Purge Isolation, CTS 3.c, is moved to improved TS 3.3.6. Improved TS 3.3.6 adds requirements on the OPERABILITY of the containment purge radiation monitors and extends the Applicability to the Manual Initiation and BOP ESFAS actuation logic to include during movement of irradiated fuel assemblies within containment and Core Alterations.	No, see CN 02-20-A.	No, see CN 2-20-A.	Yes	Yes
02-06 LS33	Functional Unit 4.a.1 of curret TS Table 3.3-3 has been deleted.	No, retained in CTS.	No, see CN 2-25-A.	No, see CN 2-25-A.	Yes
02-07 LS11	[Note (a) is added to CTS Table 3.3-3 for the Steam Line Isolation Function to state that the LCO requirements are not applicable in MODES 2 and 3 when the MSIVs are closed and deactivated]. Note [(b)] is added to CTS Table [3.3-3] for the Feedwater Isolation and Turbine Trip Function to state that the LCO requirements are not applicable when the [MFIVs, MFRVs and the associated bypass valves] are closed [and deactivated or isolated by a closed manual valve]. or & insert (1)	Yes	Yes	Yes	Yes
02-08 M	[This change revises ACTION 20 and 35 in CTS Table 3.3-3 and adds new ACTION 35.2 which are applicable to Units 1.c, 1.d, 1.f, 4.d, 4.e, 5.b, 6.c.2a, and 6.d]. These ACTION Statements, written to reflect the Applicability of the affected channels, are more restrictive, by one hour, than the current ACTION Statement[s] which invoke[ ] LCO 3.0.3 if the inoperable channel is not placed in trip within 6 hours.	Yes	Yes	Yes	Yes

(M)

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
02-29 M	A new functional unit 9 is added to the DCPP CTS, per a License Amendment Request 97-10, that incorporates ACTION ~~20.1~~ 36 (new) and the appropriate Surveillances for the residual heat removal (RHR) pump trip from low refueling water storage tank (RWST) level.	Yes	No	No	No
02-30 A	This change deletes Note (3) from Functional Unit 2.b. The slave relays listed in Note (3) were never associated with containment spray.	No, not in CTS.	No, not in CTS.	Yes	Yes
02-31 A	The *** Note for Functional Unit 6.g of CTS Table 3.3-3 is deleted. This note is no longer needed given that the ITS provideds separate Condition entry on a per pump basis, as evaluated under CN 2-46-LS42, and given the adoption of ITS LCO 3.0.6 and the Safety Function Determination Program.	No, not in CTS.	No, not in CTS.	Yes	Yes
02-32 LS23	These changes affect the ACTION STATEMENTs for inoperable undervoltage or degraded voltage relays. For all trip functions, new statements are added which require that the diesel generator, which was made inoperable by the inoperable undervoltage or degraded voltage relay, may be declared inoperable in lieu of entering LCO 3.0.3. If the number of OPERABLE channels per bus is less than the Required Number, one hour is now allowed to restore the channel to operability before compensatory actions are required.	No, see CN-02-48-LS28.	Yes	No, see CN 2-18-LS-31.	No, see CN 2-18-LS-31.

[illegible]

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-05 LS14	ACTION Statement [34] for the [Control Room Air Intake] [ ] radiation monitors have extended Completion Times, from [1 hour] to 7 days for one required channel inoperable.	Yes	Yes	Yes	Yes
03-06 A	ACTION [c] of CTS LCO 3.3.3.1 is revised to state the Specification 3.0.3 exception is [retained only for the Fuel Handling Building Radioactivity Instrumentation].	Yes	Yes	Yes	Yes
03-07 LS16	The Applicability for the Fuel [Handling] Building Exhaust radiation monitors has been revised to read "during movement of irradiated fuel assemblies in the fuel [handling] building." [The REQUIRED CHANNELS for Instrument 1.b. has been revised from one as specifed by the CTS to two as specifed by NUREG-1431 to provide single failure protection.]	Yes	No, not in CTS.	Yes	Yes
03-08 M	The DCPP CTS have been revised to include manual initiation of the fuel handling building and automatic initiation of the control room pressurization system. These systems are not classified as ESF functions in the CTS. This revision incorporates the Actuation Logic, Master Relay, and Slave Relay Tests included in NUREG-1431for the CRVS and the TADOT for the manual actuation of both systems.	Yes	No	No	No
03-09 LS24	The CPSES Surveillance frequency for the performance of a CHANNEL OPERABILITY TEST for the radiation monitoring instrumentation channels would be extended from once per 31 days to once per 92 days. This change is consistent with the ITS.	No	Yes	No	No
03-10 LG	The DCPP descriptive information related to the Required Channels per normal intake is moved to the Bases.	Yes	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-15 M	This change revises DCPP CTS ACTION 34 and adds new ACTION 36 to require appropriate MODE changes or condition changes for the CRVS with one or two inoperable normal intake monitors. These actions specify a shutdown requirement for MODES 1-4 that is one hour less than LCO 3.0.3, and immediate action for inoperability in MODE 5 or 6, and immediate action for inoperability during fuel movement. Refer also to CN 03-08-M, CN 03-04-M, and CN 3.3-51.	Yes	No	No	No
03-16 ~~A~~ Not used.	~~ITS 3.3.6 for DCPP includes MODES 1-4 and during movement of irradiated fuel assemblies within containment, in addition to MODE 6, in the LCO Applicability. These requirements are inferred in CTS 3.6.3 and are repeated here for clarity.~~	~~Yes~~ N/A	~~No~~ N/A	~~No~~ N/A	~~No~~ N/A
03-17 A	The CPSES restrictions on opening of the containment pressure relief valves is moved from the Radiation Monitoring Instrumentation specification in the CTS to ITS 3.6.3 and the ITS Administrative Controls Section 5.5.1 for the ODCM.	No	Yes	No	No
04-01 R	DCPP LCO 3.3.3.2, Movable Incore Detectors, is relocated to a licensee controlled document.	Yes, see Attachment 21, page 11.	No	No	No
05-01 R	DCPP LCO 3.3.3.3, Seismic Instrumentation, is relocated to a licensee controlled document.	Yes, see LAR 95-07	No	No	No
06-01 R	DCPP LCO 3.3.3.4, Meterological Instrumentation, is relocated to a licensee controlled document.	Yes, see Attachment 21, page 13.	No	No	No

CONVERSION COMPARISON TABLE - CURRENT TS 3/4.3

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
08-04 LS17	Consistent with NUREG-1431 Rev. 1 (ITS 3.3.3 Required Actions C.1, E.1, and G.1), this change deletes the requirement to initiate an alternate means of monitoring within 72 hours when two channels of Containment Radiation Level [~~, main steam line radiation monitors or the plant vent radiation high range~~] are inoperable as specified in C[TS 3.3.3.6 ACTION d].	Yes	Yes	Yes	Yes
08-05 A	In conjunction with CN 8-03-A, this change rearranges the ACTION Statements for the single channel Functions (i.e., SG Water Level - Wide Range and AFW Flow Rate). No change in AOT is requested, therefore this is an administrative change.	No, see CN 8-11-LS-30.	No, not in CTS.	Yes	Yes
08-06 LG	Specific equipment ID numbers are moved to the ITS Bases.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
08-07 A	This CPSES-specific change revises the definition of Core Exit Temperature thermocouple channels. The change clarifies that 2 Core Exit Temperature thermocouple channels per quadrant per train with each channel consisting of 1 thermocouple, is the same as 2 channels per quadrant with each channel consisting of 2 thermocouples.	No	Yes	No	No

(m)

(E)

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS1
(continued)

DCPP

testing requirements for those other sensors. The specific sensors installed at [~~Callaway~~] and that require RTT are listed below:

- Steam Generator Water Level	Rosemount/Model 1154
- Pressurizer Pressure	Rosemount/Model 1154
- Steamline Pressure	Barton/Model 763 and Rosemount/Model 1154
- Containment Pressure	Rosemount/Model 1154
- Reactor Coolant Flow	Rosemount/Model 1154

The basis for eliminating periodic response time testing for each sensor is discussed in the WCAP and/or the EPRI report. These reports provide justification that any sensor failure that significantly degrades response time will be detectable during surveillance testing such as calibration and channel checks.

The applicability of the generic analysis of WCAP-13632-P-A Revision 2 has been confirmed for [DCPP]. Each of the above transmitters is included in Table 9-1 of WCAP-13632. In addition, the following discussion addresses the four actions raised in the NRC SER dated September 5, 1995:

(a) A hydraulic response time test will be performed on any new or refurbished transmitter, prior to declaring the affected channel operable, to determine an initial sensor-specific response time value.

(b) A hydraulic response time test will be performed on units that use capillary tubes after initial installation of replacement transmitters or following any maintenance or modification activity that could damage the capillary tubing or degrade the response time characteristics of installed sensors.

(c) [DCPP] does not utilize pressure sensors that incorporate a variable damping feature in the RTS or ESFAS channels that are required to have their response times verified.

(d) {DCPP has established an enhanced monitoring program for those Rosemount transmitters that require monitoring per NRC Bulletin 90-01 and Supplement 1. Some of those transmitters are RTS or ESFAS transmitters that require RTT. These transmitters will remain in the enhanced monitoring program until they are replaced or the alternative method of performing periodic drift monitoring as described below is initiated:

1. Ensure that operators and technicians are aware of the Rosemount transmitter loss of fill-oil issue and make provisions to ensure that technicians monitor for for sensor response time degradation during the performance of calibrations and functional tests of these transmitters, and

2. Review and revise surveillance testing procedures, if necessary to ensure that calibrations are being performed using equipment designed to provide a step function or fast ramp in the process variable and that allows simultaneous monitoring of both the input and output response of the transmitter under test. Thus allowing, with reasonable assurance, the recognition of significant response time degradation.}

Based on these results, the Technical Specifications are revised to indicate that the system response time shall be verified utilizing a sensor response time justified by the methodology described in WCAP-13632-P-A Revision 2. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

(2)

NSHC LS2
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

A new ACTION Statement [2.1] is created which is essentially the same as current ACTION Statement [6], and is similar to current ACTION Statement [2], but does not require a reduction in THERMAL POWER to less than 75% RTP or the measurement of the QPTR if above 75% RTP. This new ACTION Statement is applied to the Power Range Neutron Flux, High Positive Rate [, High Negative Rate] trip functions. Since these are rate functions, their effectiveness is not improved by reducing power.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> 1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
> 2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
> 3. *Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the three categories of the significant hazards consideration standards, as applied to the relaxation applicable to the power range neutron flux, high positive rate [and high negative rate] trip functions:

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

   Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the ACTION associated with an inoperable channel in the power range neutron flux, high positive rate [and high negative rate] trip function. No power reduction below 75% RTP or QPTR monitoring above 75% RTP would be required since these actions have no basis for this rate trip function.

   The high positive rate [and high negative rate] trip functions are insensitive to the static power level. The proposed change will not affect any of the analysis assumptions for any of the accidents previously evaluated. [ ] The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

6

NSHC LS8
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

This change reflects a revision to current ACTION Statement [6]. If the requirements of current ACTION Statement [6] are not met, LCO 3.0.3 would be entered. This ACTION Statement is revised to state that if the ACTION requirements are not met, THERMAL POWER must be reduced to below the P-7 interlock setpoint within the next 6 hours. Most of the Functional Units that impose ACTION Statement [6], Pressurizer Pressure - Low, Pressurizer Water Level - High, Reactor Coolant Flow - Low, Two Loops (above P-7 and below P-8), RCP Undervoltage, and RCP Underfrequency) are automatically blocked below P-7 and an Applicability Note has been added accordingly. The Reactor Coolant Flow - Low (Single Loop) reactor trip function does not have to be OPERABLE below the P-8 setpoint; however, the Required Action must take the plant below the P-7 setpoint, if an inoperable channel is not tripped within 6 hours, due to the shared components between this function and the Reactor Coolant Flow - Low (Two Loops) trip function.

The proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> *1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> *2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> *3. Involve a significant reduction in a margin of safety."*

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

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

   Overall protection system performance will remain within the bounds of the previously performed accident analyses since no hardware changes are proposed. The proposed change adds a relaxation to the ACTION Statement associated with an inoperable channel in CTS Table 3.3-1 Functional Units [9, 11, 12.a, 12.b, 14, and 15] by keeping the end point of the shutdown action above the CTS requirement if an inoperable channel isn't placed in trip within 6 hours. The new ACTION Statement would reduce power to less than P-7 (10% RTP) within the next 6 hours in this situation as compared to entry into LCO 3.0.3 (power $\leq$ 5% RTP) in the current TS. The proposed change in the ACTION Statement will not affect any of the analysis assumptions for any of the accidents previously evaluated. An LCO 3.0.3 shutdown to $\leq$ 5% RTP is not required to meet the initial conditions of any accident analysis crediting these trip functions. The proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety-related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety-related equipment assumed to function during an accident situation. Therefore, the proposed

C

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
		(continued)
SR 3.3.1.14	-------------------NOTE-------------------- Verification of setpoint is not required. -- Perform TADOT.	~~18~~ 24 months B
SR 3.3.1.15	-------------------NOTE-------------------- Verification of setpoint is not required. -- Perform TADOT.	------NOTE------ Only required when not performed within previous 31 days --------------- Prior to reactor startup
SR 3.3.1.16	-------------------NOTE-------------------- Neutron detectors are excluded from response time testing. -- Verify RTS RESPONSE TIME is within limits.	~~18~~ 24 months B on a STAGGERED TEST BASIS
SR 3.3.1.17	Perform ACTUATION LOGIC TEST	3.3-45 18 months

3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. One channel inoperable.	I.1 ----------NOTE---------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. -------------------------- Place channel in trip.	B 6 hours
	OR	
	I.2.1 Be in MODE ~~3~~2.	12 hours 3.3-127
	AND	
	1.2.2 Be in MODE 3 for function 5.b	12 hours
J. NOT USED ~~One Main Feedwater Pumps trip channel inoperable.~~	~~J.1 Restore channel to OPERABLE status.~~	~~48 hours~~ 3.3-116
	~~OR~~	
	~~J.2 Be in MODE 3.~~	~~54 hours~~
K. ~~NOT USED~~ ~~One channel inoperable.~~ One channel inoperable	~~K.1.1 NOTE~~ ~~One additional channel may be bypassed for up to 4 hours for surveillance testing.~~ -------------------------- ~~Place channel in bypass.~~	3.3-29 ~~6 hours~~
	K.1.1 Place channel in cut-out.	6 hours
	AND	
	OR	
	K.1.2 Return the inoperable channel to an OPERABLE STATUS	72 hours ~~(continued)~~
	OR	
	K.2.1 Be in Mode 3	6 hours
	AND	
	K.2.2 Be in Mode 5	36 hours

(continued)

C

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.8	-------------------NOTE------------------------ Verification of setpoint not required for manual initiation functions. --- Perform TADOT.	24 ~~18~~ months B
SR 3.3.2.9	--------------------NOTE----------------------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. --- Perform CHANNEL CALIBRATION.	18 months B
SR 3.3.2.10	--------------------NOTE----------------------- Not required to be performed for the turbine driven AFW pump until 24 hours after SG pressure is ≥ ~~[1000]~~ 650 psig. --- Verify ESFAS RESPONSE TIMES are within limit.	B B-PS 24 ~~18~~ months on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.3.2.11	------------------NOTE------------------ Verification of setpoint not required. -- Perform TADOT.	~~Once per reactor trip breaker cycle~~ ~~18~~ 24 months	3.3-61
SR 3.3.2.12	Perform Actuation Logic Test	24 months	3.3-29
SR 3.3.2.13	------NOTE------ Verification of setpoint not required for manual initiation functions ---------------- Perform TADOT	18 months	3.3-139

Table 3.3.2-1 (page ~~6~~7 of ~~8~~11)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED TRIP SETPOINT~~(a)~~
5. Turbine Trip and Feedwater Isolation						PS
a. Automatic Actuation Logic and Actuation Relays	1.2$^{(j)}$, ~~[3]~~~~(j)~~	2 trains	H ~~[G]~~	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. SG Water Level – High High (P-14)	1.2$^{(j)}$, ~~[3]~~~~(j)~~	3 per SG	I ~~[D]~~	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ ~~[84.2]~~ 75.5%	B B-PS ≤ ~~[82.4]~~ 75%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
6. Auxiliary Feedwater						3.3-58
a. Manual	1.2.3	1 sw/pp	N	SR 3.3.2.~~8~~13	NA	NA 3.3-139
~~a~~ b. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1.2.3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. NOT USED ~~b. Automatic Actuation Logic and Actuation Relays (Balance of Plant ESFAS)~~	~~1.2.3~~	~~2 trains~~	G	~~SR 3.3.2.3~~	~~NA~~	3.3-01 ~~NA~~
~~c~~d. SG Water Level – Low Low	1.2.3 ~~(k)~~	3 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ ~~[30.4]~~ 6.8%	B-PS B 3.3-46 ≥ ~~[32.2]~~ 7.2%

(E)

Table 3.3.2-1 (page ~~7~~9 of ~~8~~11)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED TRIP SETPOINT~~(a)~~
6. Auxiliary Feedwater (continued)						
~~d~~e. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
						3.3-01
~~e~~f. NOT USED ~~Loss of Offsite Power~~	~~1,2,3~~	~~[3] per bus~~	~~F~~	~~SR 3.3.2.7~~ ~~SR 3.3.2.9~~ ~~SR 3.3.2.10~~	~~≥ [2912] V with ≤ 0.8 sec time delay~~	~~≥ [2975] V with ≤ 0.8 sec time delay~~
						B-PS 3.3-127
~~f~~g. Undervoltage Reactor Coolant Pump	1,~~2~~	~~[3]~~ 2 per bus	I	SR 3.3.2.~~7~~8 SR 3.3.2.9 SR 3.3.2.10	≥ ~~[69] % bus voltage~~ 7730 volts	≥ ~~[70]% bus voltage~~ 8050 volts
						3.3-116
~~g~~h. NOT USED~~Trip of all Main Feedwater Pumps~~	~~1,2~~	~~[2] per pump~~	~~J~~	~~SR 3.3.2.8~~ ~~SR 3.3.2.9~~ ~~SR 3.3.2.10~~	~~≥ [ ] psig~~	~~≥ [ ] psig~~
						3.3-01
~~h~~i. NOT USED ~~Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low~~	~~1,2,3~~	~~[2]~~	~~F~~	~~SR 3.3.2.1~~ ~~SR 3.3.2.7~~ ~~SR 3.3.2.9~~	~~≥ [20.53] [psia]~~	~~≥ [ ] [psia]~~
7. ~~Automatic Switchover to Containment Sump~~						
~~a. Automatic Actuation Logic and Actuation Relays~~	~~1,2,3,4~~	~~2 trains~~	~~C~~	~~SR 3.3.2.2~~ ~~SR 3.3.2.4~~ ~~SR 3.3.2.6~~	~~NA~~	~~NA~~

3.3-01

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	ED TRIP SETPOINT~~(a)~~
~~b. Refueling Water Storage Tank (RWST) Level - Low Low~~	~~1,2,3,4~~	4	~~K~~	~~SR 3.3.2.1~~ ~~SR 3.3.2.5~~ ~~SR 3.3.2.9~~ ~~SR 3.3.2.10~~	~~≥ [15]% and ≤ [ ]%~~	~~≥ [ ] and ≤ [ ]~~
~~Coincident with Safety Injection~~	~~Refer to Function 1 (Safety Injection) for all initiation functions and requirements.~~					3.3-29
7. Residual Heat Removal Pump Trip on Refueling Water Storage Tank Level-Low	1,2,3,4	3	K	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.12	≤ 33.68 % ≥ 31.44 %	32.56 %

(continued)

~~(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~ ED

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Source Range Neutron Flux (continued)

~~OPERABLE~~ to provide core protection against a rod withdrawal accident. If the ~~CRD~~ Rod Control System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like an uncontrolled boron dilution. ~~These inputs are provided to the BDPS.~~ The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.3, "Nuclear Instrumentation."

6. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection and it protects against vessel exit bulk boiling and ensures that the exit quality is within the limits defined by the DNBR correlation. The inputs to the Overtemperature ΔT trip include ~~all~~ pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. ~~The Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase.~~ The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature – the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;

- pressurizer pressure – the Trip Setpoint is varied to correct for changes in system pressure; and

- axial power distribution – f(ΔI), the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the

- NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

- Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

Redline

ΔT_0, as used in the overtemperature and overpower ΔT trips, represents the 100 percent RTP value of ΔT as measured by the

(continued)

BASES

plant for each loop. For the initial startup of a refueled core, ΔT_0 is initially assumed to be the same as the last measured ΔT value from the previous cycle until ΔT is measured again at full power. Accurate determination of the loop specific ΔT values should be made quarterly when performing the incore/excore recalibration at steady-state conditions (i.e., power distributions not affected by xenon or other transient conditions). The variation in indicated ΔT between loops is due to the variance in both real hot leg temperatures and hot leg temperature measurement biases. The real hot leg temperature variance between loops is primarily caused by asymmetrical flow in the upper plenum, and the difference in hot leg temperature measurement bases, primarily caused by differences in hot leg temperature streaming error between loops. The change in the indicated loop ΔTs with burn up is caused primarily by the change in the hot leg streaming biases as the radial power distribution changes.

Retain

D

The Overtemperature ΔT trip Function is calculated for each loop as described in Note 1 of Table 3.3.1-1. Trip occurs if Overtemperature ΔT is indicated in two loops. ~~At some units,~~ The pressure and temperature signals are used for other control functions~~.~~; ~~For those units,~~ thus the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE ~~for two and four loop units (the LCO requires all three channels on the Overtemperature ΔT trip Function to be OPERABLE for three loop units)~~. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB (2-out-of-4 coincidence). In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

(continued)

D & E

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

7. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions for Condition I and II event (Ref. 12). This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. The overpower ΔT trip also provides protection to mitigate the consequences of small steamline breaks, as reported in WCAP-9226, Ref. 16, and steamline breaks with coincident control rod withdrawal (Ref. 3). It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and
- rate of change of reactor coolant average temperature - including dynamic compensation for the delays between the core and the temperature measurement system.

Red-line this text

ΔT_0, as used in the overtemperature and overpower ΔT trips, represents the 100 percent RTP value of ΔT as measured by the plant for each loop. For the initial startup of a refueled core, ΔT_0 is initially assumed to be the same as the last measured ΔT value from the previous cycle until ΔT is measured again at full power. Accurate determination of the loop specific ΔT values should be made quarterly when performing the incore/excore recalibration at steady-state conditions (i.e., power distributions not affected by xenon or other transient conditions). The variation in indicated ΔT between loops is due to the variance in both real hot leg temperatures and hot leg temperature measurement biases. The real hot leg temperature variance between loops is primarily caused by asymmetrical flow in the upper plenum, and the difference in hot leg temperature measurement biases ins primarily caused by differences in hot leg temperature streaming error between loops. The change in the indicated loop ΔTs with burn up is caused primarily by the change in the hot leg streaming biases as the radial power distribution changes.

The Overpower ΔT trip Function is calculated for each loop as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two loops. ~~At some units.~~ The temperature signals are used for other control functions~~.~~; ~~At those units.~~ thus the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip. The LCO requires four channels ~~for two and four loop units (three channels for three loop units)~~ of the Overpower ΔT trip Function to be OPERABLE (2-out-of-4 coincidence). Note that the Overpower ΔT trip Function receives input from ~~consequences of small steamline breaks, as reported in WCAP 9226, Ref. 6, and steamline breaks with coincident control rod withdrawal (Ref. 3).~~

delete redline

add strike-out (this text belongs in first paragraph on page B.3.3-17)

6 16 3

(continued)

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

should be redline

The same sensors provide input to the Pressurizer Pressure – High and – Low trips and the Overtemperature ΔT trip. ~~At some units.~~ The Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System~~.~~ ; ~~For those units~~ thus ~~the~~ the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

a. Pressurizer Pressure – Low

The Pressurizer Pressure – Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

The LCO requires four channels ~~for two and four loop units (three channels for three loop units)~~ of Pressurizer Pressure – Low to be OPERABLE (2-out-of-4 coincidence).

In MODE 1, when DNB is a major concern, the Pressurizer Pressure – Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 Low Pressure Permissive interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent

(continued)

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

11. Reactor Coolant Pump (RCP) Breaker Position

~~Both RCP Breaker Position trip Functions operate together on two sets of auxiliary contacts, with one set on each RCP breaker. These Functions anticipate the Reactor Coolant Flow—Low trips to avoid RCS heatup that would occur before the low flow trip actuates.~~

add strike-out

a. ~~Reactor Coolant Pump Breaker Position (Single Loop)~~
NOT USED

~~The RCP Breaker Position (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open above the P-8 setpoint, a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low (Single Loop) Trip Setpoint is reached.~~

~~The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE. One OPERABLE channel is sufficient for this trip Function because the RCS Flow—Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.~~

~~This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.~~

~~In MODE 1 above the P-8 setpoint, when a loss of flow in any RCS loop could result in DNB conditions in the core, the RCP Breaker Position (Single Loop) trip must be OPERABLE. In MODE 1 below the P-8 setpoint, a loss of flow in two or more loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR.~~

(continued)

add strike-out

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~b. Reactor Coolant Pump Breaker Position (Two Loops)~~

The RCP Breaker Position ~~(Two Loops)~~ trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The position of each RCP breaker is monitored. Above the P-7 setpoint ~~and below the P-8 setpoint~~, a loss of flow in two or more loops will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low ~~(Two Loops)~~ Trip Setpoint is reached.

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE (2-out-of-4 coincidence). One OPERABLE channel is sufficient for this Function because the RCS Flow-Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-7 setpoint ~~and below the P-8 setpoint~~, the RCP Breaker Position ~~(Two Loops)~~ trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since ~~no conceivable power distributions could occur that would cause a DNB concern at this low power level~~there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in two RCS loops is automatically enabled. ~~Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.~~

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E D

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

12. Undervoltage Reactor Coolant Pumps

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in ~~two or more~~all RCS loops. The voltage to ~~each~~both RCP buses is monitored by two relays each. Above the P-7 setpoint, a loss of voltage detected on ~~two or more~~both RCP buses, i.e. a complete loss of flow event, will initiate a reactor trip. ~~This~~For this event, the under voltage trip Function will generate a reactor trip before the Reactor Coolant Flow—Low ~~(Two Loops)~~ Trip Setpoint is reached. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires ~~three~~ two Undervoltage RCPs channels ~~(one per phase)~~ per bus to be OPERABLE (1 per bus both busses).

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked ~~since no conceivable power distributions could occur that would cause a DNB concern at this low power level~~, since there is insufficent heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in ~~two or more of~~ four RCS loops is automatically enabled. ~~This Function uses the same relays as the ESFAS Function 6.f, "Undervoltage Reactor Coolant Pump (RCP)" start of the auxiliary feedwater (AFW) pumps.~~

Strike-out

All

13. Underfrequency Reactor Coolant Pumps

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. ~~The proper~~ An adequate coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 setpoint, a loss of frequency detected ~~on two or more~~ by two relays on one RCP buse~~s~~ will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low ~~(Two Loops)~~ Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

(continued)

remove Strike out (Three is correct)

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

13. Underfrequency Reactor Coolant Pumps (continued)

The LCO requires ~~three~~ two Underfrequency RCPs channels per bus to be OPERABLE ~~(1 per bus both busses)~~.

In MODE 1 above the P-7 setpoint, the Underfrequency RCPs trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked ~~since no conceivable power distributions could occur that would cause a DNB concern at this low power level~~since there is insufficient heat production to be concerned about DNB. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

14. Steam Generator Water Level – Low Low

The SG Water Level – Low Low trip Function ensures that protection is provided against a loss of heat sink ~~and actuates the AFW System prior to uncovering the SG tubes~~ in the event of a loss of feedwater flow to one or more SGs. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This Function also performs the ESFAS function of starting the AFW pumps on low low SG level.

The LCO requires ~~four~~ three channels of SG Water Level – Low Low per SG (1 per SG in one SG)and four channels of RCS ΔT (1/loop) to be OPERABLE. The installation of the median signal selector (MSS) and four channels of RCS ΔT (1/loop) effectively eliminates the possibility that a single random failure could cause a control system action that results in a condition requiring protection action, and also prevent proper operation of a protection system channel designed to protect against the condition. Thus, the MSS prevents interaction between the feedwater control and reactor protection systems in accordance with the requirements of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." Removal of this interaction eliminates the need for the low feedwater flow reactor trip. The MSS will functionally separate steam generator narrow range level protection channels (low-low steam generator water level trip) to provide compliance with IEEE 279-1971 and satisfy the original design basis. ~~for four loop units in which these channels are shared between protection and control. In two, three, and four loop units where three SG Water Levels are dedicated to the RTS, only three channels per SG are required to be OPERABLE.~~ This trip is actuated on two out of three low-low water level signals occurring in any steam generator. If a low-low water level condition is detected in one steam generator, signals shall be generated to trip the reactor and start the motor

(continued)

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Turbine Trip—Turbine Stop Valve Closure (continued)

a load rejection, and the Turbine Trip—Stop Valve Closure trip Function does not need to be OPERABLE.

17. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for the small break LOCA but rod insertion is not credited for the large break LOCA (Ref. 3). However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoint and Allowable Values are not applicable to this Function. The SI Input is provided by ~~relay~~ logic in the SSPS circuitry of ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2 (1-out-of-2 coincidence).

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function ~~must be OPERABLE in MODE 1 or 2,~~ when the reactor is critical, ~~and must be shut down in the event of an accident~~. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE. e

18. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE

(continued)

E

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

18. Reactor Trip System Interlocks (continued)

when the associated reactor trip functions are outside the applicable MODES. These are:

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following
functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip and allows the high voltage to be de-energized. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. and ~~When the source range trip is blocked, the high voltage to the detectors is also removed;~~

- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip. ~~and~~ (insert period)

- ~~on increasing power, the P-6 interlock provides a backup block signal to the source range flux doubling circuit. Normally, this Function is manually blocked by the control room operator during the reactor startup.~~

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint (1-out-of-2 coincidence).

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

c. Power Range Neutron Flux, P-8 (continued)

(E)

power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires ~~four~~ three channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1 (2-out-of-3 coincidence).

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9

The Power Range Neutron Flux, P-9 interlock is actuated at ~~approximately~~ less than or equal to 50% power as determined by two-out-of-four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip—Low ~~Fluid~~ Auto Stop Oil Pressure and Turbine Trip—Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip ~~will~~ may challenge the pressurizer PORVs due to the mismatch between reator power and ~~cause a load rejection beyond the capacity~~ capacities of the Steam Dump and Reactor Control Systems. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

The LCO requires ~~four~~three channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1 (2-out-of-3 coincidence).

~~In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE.~~ In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a significant load rejection ~~beyond the capacity capacities of the Steam Dump and Reactor Control Systems.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at ~~approximately~~ 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal;
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux—Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a back up signal to block the Source Range Neutron Flux ~~reactor trip, and also to de-energize the NIS source range~~ detectors high voltage and allows manual block of the IR rod stop; remove strike-out
- the P-10 interlock provides one of the two inputs to the P-7 interlock; ~~and~~
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux—Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop)~~.~~; and
- on decreasing power, the P-10 interlock automatically defeats the block of the source range neutron flux trip and with P-6 energizes the source range high voltage.

The LCO requires ~~four~~three channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2(2-out-of-3).

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a

(continued)

B

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

21. Automatic Trip Logic (continued)

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when ~~the RTBs and associated bypass breakers are closed, and~~ the ~~CRD~~ Control Rod System is capable of rod withdrawal or all rods are not fully inserted.

22. Seismic Trip

The seismic trip system operates to shut down reactor operations should ground accelerations exceed a preset level in any two of the three orthogonal directions monitored (one vertical, two horizontal).

Three triaxial sensors (accelerometers) are anchored to the containment base in three separate locations 120 degrees apart. Each senses acceleration in three mutually orthogonal directions. Output signals are generated when ground accelerations exceed the preset level. These signals are transmitted to the Trains A and B Solid State Protection System (SSPS). If two of the three sensors in any direction produce simultaneous outputs, the logic produces trains A and B reactor trip signals.

This trip function must be OPERABLE in MODE 1 or 2 when the reactor is critical and must be capable to shut down the reactor in the event of an earthquake. Three channels in these directions are required to be OPERABLE.

The RTS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii) ~~the NRC Policy Statement.~~

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. When the Required Channels in Table 3.3.1-1 are specified (e.g., on a per steam line, per loop, per basis), then the Condition may be entered separately for each steam line, loop, SG, etc., as appropriate.

Insert 3.3.1 Actions

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

~~Reviewer's Note: Certain LCO Completion Times are based on approved topical reports. In order for a licensee to use these times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.~~

(continued)

(B)

In the event a channel's setpoint is found nonconservative with respect to the specified Trip Setpoint, but more conservative than the Allowable Value, the setpoint must be adjusted consistent with the Trip Setpoint value. When a channel's Trip Setpoint is nonconservative with respect to the Allowable Value, declare the channel inoperable and apply the applicable ACTION statement until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

INSERT 3.3.1 ACTIONS

(A)

BASES

ACTIONS

G.1 and G.2 (continued)

level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

H.1 – Not used

~~Condition H applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is below the P-6 setpoint and one or two channels are inoperable. Below the P-6 setpoint, the NIS source range performs the monitoring and protection functions. The inoperable NIS intermediate range channel(s) must be returned to OPERABLE status prior to increasing power above the P-6 setpoint. The NIS intermediate range channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10.~~

I.1

Condition I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

J.1

Condition J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the ~~RTBs closed and the CRD~~ Rod Control System capable of rod withdrawal or all rods not fully inserted. With the unit in this Condition, below P-6, the

(continued)

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BASES

ACTIONS

S.1 and S.2 (continued)

within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

T.1 and T.2

Condition T applies to the P-7, P-8, P-9, and P-13 interlocks. With one or more required channel(s) inoperable, ~~for one-out of two or two-out of four coincidence logic,~~ the associated interlock must be verified by observation of the associated permissive annunciator window to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

U.1~~, U.2.1,~~ and U.2~~.2~~

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time) ~~followed by opening the RTBs in 1 additional hour (55 hours total time)~~. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

(continued)

A

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.6 (continued)

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > ~~50~~ 75% RTP and that ~~[24]~~ hours after achieving equilibrium conditions with thermal power ≥75% RTP is allowed for performing the first surveillance ~~after reaching 50 75% RTP~~.

space

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every ~~[92]~~ days.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function.

Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

~~The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology (trip setpoint value).~~

delete redline

~~The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of Reference 7.~~

SR 3.3.1.7 is modified by two notes ~~a~~ Note 1 ~~that~~ provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3. Note 2 requires that the quarterly COT for the source range instrumentation shall include verification by observation of the associated permissive annunciator window that the P-6 and P-10 interlocks are in their required state for the existing unit conditions.

The Frequency of ~~[92]~~ days is justified in Reference 7.

(continued)

C

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.13

SR 3.3.1.13 is the performance of a COT of RTS interlocks every ~~[18]~~ months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.14

24

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, Seismic Trip and the SI Input from ESFAS. This TADOT is performed every ~~[24]~~ months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.15

SR 3.3.1.15 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to reactor startup. A Note states that this Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.

SR 3.3.1.16

SR 3.3.1.16 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in ~~Technical Requirements Manual, Section 15 (Ref. 8)~~ the FSAR (Ref. 1). Individual component response times are not modeled in the analyses.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.16 (continued)

The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer Function set to one, with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value, provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

The response time testing for the SG water level low-low does not include trip time delays. Response times include the transmitters, Eagle-21 process protection cabinets, solid state protection system cabinets, and actuation devices only. This reflects the response times necessary for THERMAL POWER in excess of 50 percent RTP. For those functions without a specified response time, SR 3.3.1.16 is not applicable.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" (Ref. 8) provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocations for sensor initial response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

As appropriate, each channel's response time must be verified every ~~[18]~~ 24 months on a STAGGERED TEST BASIS. Each verification shall include at least one train such that both trains are verified at least once per ~~36~~ 48 months. Testing of the final actuation devices is included in the ~~testing~~verification. Response times cannot be

(continued)

BASES

14. WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals", April 1994

15. WCAP-14117, "Reliability Assessment of Potter and Brumfield MDR Series Relays."

16. WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases," Revision 1, January 1978.

17. WCAP-11082, Rev. 5, "Westinghouse Setpoint Methodology for Protection Systems, Diablo Canyon Units 1 and 2, 24 month Fuel Cycle Evaluation," January 1997.

(B)

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND

The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into three distinct but interconnected modules as identified below:

- Field transmitters or process sensors and instrumentation: provide a measurable electronic signal based on the physical characteristics of the parameter being measured;
- Signal processing equipment including ~~analog~~ digital protection system, field contacts, and protection channel sets: provide signal conditioning, bistable setpoint comparison, process algorithm actuation, compatible electrical signal output to protection system devices, and control board/control room/ miscellaneous indications; and
- Solid State Protection System (SSPS) including input, logic, and output bays: initiates the proper unit shutdown or engineered safety feature (ESF) actuation in accordance with the defined logic and based on the bistable outputs from the signal process control and protection system.

The residual heat removal pump trip or refueling water storage tank level-low is not processed by the SSPS. The associated relays are located in the residual heat removal pump control system.

Field Transmitters or Sensors

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). In some cases, the same channels also provide control system inputs. To account for calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the Trip Setpoint and Allowable

(continued)

B

BASES

BACKGROUND

Signal Processing Equipment (continued)

actuation. Again, a single failure will neither cause nor prevent the protection function actuation.

These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 2.

The channels are designed such that testing required to be performed at power may be accomplished without causing an ESF actuation.

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

The Trip Setpoints used in the bistables are based on the analytical limits stated in Reference 2. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.2-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in ~~the "RTS/ESFAS Setpoint Methodology Study" Study"~~ WCAP-11082, Rev. 2, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Station - Eagle 21 Version," May 1993(Ref. 6). The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

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Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

The Process Protection System is designed to permit any one channel to be tested and maintained at power in a bypassed mode. If a channel has been bypassed for any purpose, the bypass is continuously indicated in the control room as required by applicable codes and standards. As an alternative to testing in the bypass mode, testing in the trip mode is also possible and permitted.

(continued)

D

BASES

BACKGROUND

Solid State Protection System (continued)

transients. If a required logic matrix combination is completed, the system will send actuation signals via master and slave relays to those components whose aggregate Function best serves to alleviate the condition and restore the unit to a safe condition. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Each SSPS train has a built in testing device that can automatically test the decision logic matrix functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

The actuation of ESF components is accomplished through master and slave relays. The SSPS energizes the master relays appropriate for the condition of the unit. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices. The master and slave relays are routinely tested to ensure operation. The test of the master relays energizes the relay, which then operates the contacts and applies a low voltage to the associated slave relays. The low voltage is not sufficient to actuate the slave relays but only demonstrates signal path continuity. The SLAVE RELAY TEST actuates the devices if their operation will not interfere with continued unit operation. For the latter case, actual component operation is prevented by the SLAVE RELAY TEST circuit, and slave relay contact operation is verified by a continuity check of the circuit containing the slave relay: ~~For slay relays in the ESF actuation system circuit that are Potter & Brumfield type MAR relays,~~ The SLAVE RELAY TEST is performed on a refueling frequency. The test frequency is based on relay reliability assessments presented in WCAP-13878, "Reliability Assessment of Potter and Brumfield MDR Series Relays," WCAP-13900, "Extension of Slave Relay Surveillance Test Intervals," and WCAP-14117, "Reliability Assessment of Potter and Brumfield MDR Series Relay." These reliability assessments are relay specific and apply only to Potter and Brumfield MDR series relays which are the only relays used in the ESF actuation system. Note that for normally energized applications, the relays may have to be replaced periodically in accordance with the guidance given in WCAP-13878 for MDR relays.

~~Reviewer's Note: No one unit ESFAS incorporates all of the Functions listed in Table 3.3.2-1. In some cases (e.g., Containment Pressure—High 3, Function 2.c), the table reflects several different implementations of the same Function. Typically, only one of these implementations are used at any specific unit.~~

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. ~~For example, Pressurizer Pressure—Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment.~~ Functions such as manual initiation, not specifically credited in the accident safety analysis, are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3).

The LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the cut-out, two-out-of-four configurations allow one channel to be tripped or bypassed during maintenance or testing without causing an ESFAS initiation. Two logic or manual initiation channels are required to ensure no single random failure disables the ESFAS.

The required channels of ESFAS instrumentation provide unit protection in the event of any of the analyzed accidents. ESFAS protection functions are as follows:

1. <u>Safety Injection</u>

   Safety Injection (SI) provides two primary functions:

   1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to < 2200°F); and

(continued)

4

BASES

- Start of AFW to ensure secondary side cooling capability;
- ~~Isolation of the control room to ensure habitability; and~~
- Transfer of the control room ventilation to ensure habitability;
- Transfer of the auxiliary building ventilation to ensure ventilation cooling to the ESF pump rooms;

(continued)

E

BASES

APPLICABLE SAFETY ANALYSES, LCO, and

a. <u>Containment Spray—Manual Initiation</u> (continued)

simultaneously and an SI signal must be present to initiate APPLICABILITY containment spray. ~~There are two sets of two switches each in the control room.~~ Simultaneously turning the two switches in either set will actuate containment spray in both trains in the same manner as the automatic actuation signal. Two Manual Initiation switches in each train are required to be OPERABLE.~~ to ensure no single failure disables the Manual Initiation Function.~~ Note that Manual Initiation of containment spray also actuates Phase B containment isolation and CVI.

b. <u>Containment Spray—Automatic Actuation Logic and Actuation Relays</u>

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of containment spray must be OPERABLE in MODES 1, 2, 3 and ~~3~~ 4 when there is a potential for an accident to occur, and sufficient energy in the primary or secondary systems to pose a threat to containment integrity due to overpressure conditions. Manual initiation is also required in MODE 4, even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated on a containment spray, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary and secondary systems to result in containment overpressure. In MODES 5 and 6, there is also adequate time for the operators to evaluate unit conditions and respond, to mitigate the consequences of abnormal conditions by manually starting individual components.

(continued)

E

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Containment Isolation - Phase A Isolation

(1) Phase A Isolation - Manual Initiation

Manual Phase A Containment Isolation is actuated by either of two switches in the control room. Either switch actuates both trains. Note that manual initiation of Phase A Containment Isolation also actuates Containment ~~Purge~~ Ventilation Isolation.

(2) Phase A Isolation - Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of Phase A Containment Isolation must be OPERABLE in MODES 1, 2, 3 and ~~3~~ 4, when there is a potential for an accident to occur. Manual initiation is also required in MODE 4 even though automatic actuation is not required. In this MODE, adequate time is available to manually actuate required components in the event of a DBA, but because of the large number of components actuated on a Phase A Containment Isolation, actuation is simplified by the use of the manual actuation ~~push buttons~~ control switches. Automatic actuation logic and actuation relays must be OPERABLE in MODE 4 to support system level manual initiation. In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Phase A Containment Isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase A Isolation - Safety Injection

Phase A Containment Isolation is also initiated by all Functions that initiate SI. The Phase A Containment Isolation

(continued)

D

BASES

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

(1) ~~Phase B Isolation—Manual Initiation~~

(2) Phase B Isolation—Automatic Actuation Logic and Actuation Relays (continued)

isolation. There also is adequate time for the operator to evaluate unit conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

(3) Phase B Isolation—Containment Pressure

The basis for containment pressure MODE applicability is as discussed for ESFAS Function 2.c above.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Rapid isolation of the steam lines will limit the steam break accident to the blowdown from one SG, at most. For an SLB upstream of the main steam isolation valves (MSIVs), inside or outside of containment, closure of the MSIVs limits the accident to the blowdown from only the affected SG. For an SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. ~~For units that do not have steam line check valves, Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.~~

a. Steam Line Isolation—Manual Initiation

Manual initiation of Steam Line Isolation can be accomplished from the control room via an individual switch on each valve. ~~There are two switches in the control room and either switch can initiate action to immediately close all MSIVs.~~ The LCO requires ~~two~~ one channel~~s~~ per valve to be OPERABLE.

(E)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Steam Line Isolation – Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and ~~[de-activated]~~ i.e., actions are taken to ensure the valves cannot be inadvertently opened. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation – Containment Pressure – High ~~2~~ High

This Function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment ~~to maintain at least one unfaulted SG as a heat sink for the reactor, and~~ to limit the mass and energy release to containment to a single SG. The transmitters (d/p cells) are located outside containment with the sensing line (high pressure side of the transmitter) located inside containment. Containment Pressure – High ~~2~~ High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. However, for enhanced reliability, this Function was designed with four channels and a two-out-of-four logic. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions, and the Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure – High ~~2~~ High must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe

(continued)

i.e., actions are taken to ensure the valves cannot be inadvertently opened

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

(1) Steam Line Pressure – Low (continued)

and 3 unless all MSIVs are closed and ~~{de-activated}~~. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

(2) Steam Line Pressure – Negative Rate – High

Steam Line Pressure – Negative Rate – High provides closure of the MSIVs for an SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure – Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure – Negative Rate – High signal is automatically enabled. Steam Line Pressure – Negative Rate – High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy requirements with a two-out-of-three logic on each steam line.

Steam Line Pressure – Negative Rate – High must be OPERABLE in MODE 3 {2 per steam line} when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure – Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MSIVs are closed and ~~{de-activated}~~. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

trip coincidence

IN ANY steam line

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, and to stop the excessive flow of feedwater into the SGs. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

The Function is actuated when the level in any SG exceeds the high high setpoint, and performs the following functions:

- Trips the main turbine;
- Trips the MFW pumps;
- Initiates feedwater isolation; and
- Shuts the MFW regulating valves and the bypass feedwater regulating valves coincident with P-4. or by an SI signal

This Function is actuated by SG Water Level – High High. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated.

a. Turbine Trip and Feedwater Isolation – Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Turbine Trip and Feedwater Isolation – Steam Generator Water Level – High High (P-14)

This signal provides protection against excessive feedwater flow. The ESFAS SG water level

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In the event of SI, the unit is taken off line and the turbine generator must be tripped. The MFW system is also taken out of operation and the AFW system is automatically started. The SI signal was discussed previously.

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BASES

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BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

h ~~i~~. ~~Auxiliary Feedwater - Pump Suction Transfer on Suction Pressure - Low (continued)~~

~~conditions and the Trip Setpoint reflects only steady state instrument uncertainties.~~

~~This Function must be OPERABLE in MODES 1, 2, and 3 to ensure a safety grade supply of water for the AFW System to maintain the SGs as the heat sink for the reactor. This Function does not have to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink. In MODE 4, AFW automatic suction transfer does not need to be OPERABLE because RHR will already be in operation, or sufficient time is available to place RHR in operation, to remove decay heat.~~

INSERT A

7. ~~Automatic Switchover to Containment Sump~~ Residual Heat Removal (RHR) Pump Trip - Refueling Water Storage Tank (RWST) Low Level

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is ~~automatically~~ manually switched to the containment recirculation sump. This pump trip feature is blocked if the RHR pumps are already taking suction from the Containment Recirculation sump. The low head residual heat removal (RHR) pumps ~~and containment spray pumps~~ draw the water from the containment recirculation sump, the RHR pumps pump the water through the RHR heat exchanger, inject the water back into the RCS, and supply the cooled water to the other ECCS pumps. Switchover from the RWST to the containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

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(B)

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7. <u>Residual Heat Removal Pump Trip on Refueling Water Storage Tank Level - Low</u>

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is manually switched to the containment recirculation sump. This pump trip feature is blocked if the RHR pumps are already taking suction from the containment recirculation sump. The low head RHR pumps draw the water from the containment recirculation sump, the RHR pumps pump the water through the RHR heat exchanger, inject the water back into the RCS, and supply the cooled water to the other ECCS pumps. Switchover from the RWST to the containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support RHR pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. The RHR pump trip on RWST low level provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with three level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-three logic is adequate to initiate the protection function actuation.

The Allowable Value/Trip Setpoint upper limit is selected to ensure adequate water inventory in the containment sump to provide RHR pump suction. The high limit also ensures enough borated water is injected to ensure the reactor remains shut down.

The transmitters are located in an area not affected by HELBs or post accident high radiation. Thus, they will not experience any adverse environmental conditions and the trip setpoint reflects only steady state instrument uncertainties.

(B)

Insert A (continued)

This Function must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. This Function is not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are prevented from actuating to prevent inadvertent overpressurization of unit systems or are not required to be operable.

A

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~a.~~ ~~Automatic Switchover to Containment Sump—Automatic Actuation Logic and Actuation Relays~~

~~Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.~~

b. ~~c.~~ ~~Automatic Switchover to Containment Sump—Refueling Water Storage Tank (RWST) Level—Low Low Coincident With Safety Injection and Coincident With Containment Sump Level—High~~

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low ~~low~~ level in the RWST ~~coincident with an SI signal~~ provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with ~~four~~ three level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-three~~four~~ logic is adequate to initiate the protection function actuation. ~~Although only three channels would be sufficient, a fourth channel has been added for increased reliability.~~

The RWST—Low ~~Low~~ Allowable Value/Trip Setpoint ~~has both upper and lower limits. The lower limit is selected to ensure switchover occurs before the RWST empties, to prevent ECCS pump damage. The~~ upper limit is selected to ensure enough borated water is injected to ensure the reactor remains shut down. The high limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction.

The transmitters are located in an area not affected by HELBs or post accident high radiation. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

~~Automatic switchover occurs only if the RWST low low level signal is coincident with SI. This prevents accidental switchover during normal~~

Strike-out all & delete all redline information

(continued)

A

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~b. c.~~ ~~Automatic Switchover to Containment Sump - Refueling Water Storage Tank (RWST) Level - Low Low Coincident With Safety Injection and Coincident With Containment Sump Level - High (continued)~~

~~These~~ This Functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. ~~These~~ This Functions ~~are~~ is not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

strike-out all & delete all redline information

8. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

a. Engineered Safety Feature Actuation System Interlocks - Reactor Trip, P-4

The P-4 interlock is enabled when a reactor trip breaker (RTB) and its associated bypass breaker is open. ~~Once the P-4 interlock is enabled, automatic SI initiation is blocked after a [ ] second time delay.~~ This Function allows operators to ~~take manual control~~ manually block reactuation of SI systems after the initial phase of injection is complete. Once SI is blocked, automatic actuation of SI cannot occur until the RTBs have been manually closed. The functions of the P-4 interlock are:

(continued)

(E)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Engineered Safety Feature Actuation System Interlocks – Pressurizer Pressure, P-11 (continued)

5, or 6 because system pressure must already be below the P-11 setpoint for the requirements of the heatup and cooldown curves to be met.

With an ESFAS Instrumentation Channel or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Values column of Table 3.3.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

c. NOT USED ~~Engineered Safety Feature Actuation System Interlocks – T_{avg} – Low Low, P-12~~

~~On increasing reactor coolant temperature, the P-12 interlock reinstates SI on High Steam Flow Coincident With Steam Line Pressure – Low or Coincident With T_{avg} – Low Low and provides an arming signal to the Steam Dump System. On decreasing reactor coolant temperature, the P-12 interlock allows the operator to manually block SI on High Steam Flow Coincident With Steam Line Pressure – Low or Coincident with T_{avg} – Low Low. On a decreasing temperature, the P-12 interlock also removes the arming signal to the Steam Dump System to prevent an excessive cooldown of the RCS due to a malfunctioning Steam Dump System.~~

~~Since T_{avg} is used as an indication of bulk RCS temperature, this Function meets redundancy requirements with one OPERABLE channel in each loop. In three loop units, these channels are used in two-out-of-three logic. In four loop units, they are used in two-out-of-four logic.~~

~~This Function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to have an accident.~~

The ESFAS instrumentation satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument Loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s)

(continued)

E

BASES

ACTIONS

C.1, C.2.1 and C.2.2 (continued)

- Phase B Isolation~~; and~~
- ~~Automatic Switchover to Containment Sump.~~

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within an additional 6 hours (12 hours total time) and in MODE 5 within an additional 30 hours (42 hours total time). The Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

The Required Actions are modified by a Note that allows one train to be bypassed for up to ~~[~~4~~]~~ hours for surveillance testing, provided the other train is OPERABLE. This allowance is based on the reliability analysis assumption of WCAP-10271-P-A (Ref. 8) that 4 hours is the average time required to perform channel surveillance.

D.1, D.2.1, and D.2.2

Remove strike-out (retain function)

Condition D applies to:

- SI- ~~Containment Pressure—High~~ ~~1;~~
- SI- Pressurizer Pressure—Low ~~(two, three, and four loop units)~~;
- SI- Steam Line Pressure—Low;
- ~~Steam Line Differential Pressure—High;~~
- ~~High Steam Flow in Two Steam Lines Coincident With T_{avg}—Low Low or Coincident With Steam Line Pressure—Low;~~

(continued)

E

BASES

ACTIONS

D.1, D.2.1, and D.2.2 (continued)

- ~~Containment Pressure — High 2;~~ High - High (add strike-out)
- Steam Line Isolation Steam Line Pressure – Negative Rate – High;
- Steam Line Pressure – Low;
- ~~High Steam Flow Coincident With Safety Injection Coincident With T_{avg} — Low Low;~~
- ~~High High Steam Flow Coincident With Safety Injection;~~
- ~~High Steam~~ Flow in Two Steam Lines Coincident ~~With T_{avg} — Low Low~~; (add strike-out)
- Auxiliary Feedwater SG Water level – Low Low ~~(two, three, and four loop units)~~; ~~and~~
- ~~SG Water level — High High (P-14) (two, three, and four loop units)~~.

If one channel is inoperable, 6 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic (excluding pressurizer pressure - low and containment pressure high - high). Therefore, failure of one channel places the Function in a two-out-of-two configuration. ~~One~~ The inoperable channel must be tripped to place the Function in a one-out-of-~~three~~two configuration that satisfies redundancy requirements.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel or one additional channel to be bypassed for up to [4] hours for surveillance testing of other channels. The 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 4 hours allowed for testing, are justified in Reference 8.

no strik-out on 4

(continued)

(E)

BASES

ACTIONS (continued)

E.1, E.2.1, and E.2.2

Condition E applies to:

- ~~Containment Pressure - High~~ 2

- Containment Spray Containment Pressure ~~- High 3~~ ~~(~~High, High~~)~~ ~~(two, three, and four loop units)~~; ~~and~~

- Containment Phase B Isolation Containment Pressure - ~~High 3~~ (High, High) ~~or~~ ; and

- Steam Line Isolation Containment Pressure - High-High

None of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray. The containment spray signal is also interlocked with SI and will not initiate without simultaneous SI and ~~CSAS~~ containment spray signals.

To avoid the inadvertent actuation of containment spray and Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 6 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status, or place it in the bypassed condition within 6 hours, requires the unit be placed in MODE 3 within the following 6 hours ~~and MODE 4 within the next 6 hours~~ and MODE 5 within 42 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4 5, these Functions are no longer required OPERABLE.

ACTIONS

(continued)

(E)

BASES

ACTIONS (continued)

G.1, G.2.1 and G.2.2

Condition G applies to the automatic actuation logic and actuation relays for the Steam Line Isolation ~~[, Turbine Trip and Feedwater Isolation,]~~ and AFW actuation Functions.

The action addresses the train orientation of the SSPS and the master and slave relays for these functions. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 8) assumption that 4 hours is the average time required to perform channel surveillance.

H.1 and H.2

Condition H applies to the automatic actuation logic and actuation relays for the Turbine Trip and Feedwater Isolation Function.

This action addresses the train orientation of the SSPS and the master and slave relays for this Function. If one train is inoperable, 6 hours are allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the following 6 hours. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of

(continued)

E

BASES

ACTIONS H.1 and H.2 (continued)

an event occurring during this interval. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. These Functions are no longer required in MODE 3. Placing the unit in MODE 3 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to ~~[4]~~ hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. 8) assumption that 4 hours is the average time required to perform channel surveillance.

I.1 and I.2

Condition I applies to:

- Turbine Trip and Feedwater Isolation — SG Water Level - High High (P-14) ~~(two, three, and four loop units)~~; and
- Auxiliary Feedwater — Undervoltage Reactor Coolant Pump.

If one channel of SG water level is inoperable, 6 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the Function is then in a partial trip condition where one-out-of-two or one-out-of-three logic will result in actuation. The 6 hour Completion Time is justified in Reference 8. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 6 hours requires the unit to be placed in MODE 3 within the following 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, ~~these~~this Function~~s~~ ~~are~~is no longer required OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, ~~these~~this Function~~s~~ ~~are~~is no longer required OPERABLE.

If one channel of undervoltage reactor coolant pump is inoperable, 6 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the function is then in a

(continued)

BASES

ACTIONS

I.1 and I.2 (continued)

partial trip condition where one additional tripped channel will result in actuation. The 6 hour Completion Time is justified in Ref. 8. Failure to restore the inoperable channel it OPERABLE status or place it in the tripped condition withing 6 hours requires the Unit to be placed in MODE 2 with in the following 6 hours. The allowed Completion time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner without challenging unit systems. In MODE 2, this Function is no longer required OPERABLE.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 6 hours allowed to place the inoperable channel in the tripped condition, and the 4 hours allowed for a second channel to be in the bypassed condition for testing, are justified in Reference 8.

J.1 and J.2

NOT USED

~~Condition J applies to the AFW pump start on trip of all MFW pumps.~~

~~This action addresses the train orientation of the SSPS for the auto start function of the AFW System on loss of all MFW pumps. The OPERABILITY of the AFW System must be assured by allowing automatic start of the AFW System pumps. If a channel is inoperable, 48 hours are allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours are allowed to place the unit in MODE 3. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above. The allowance of 48 hours to return the train to an OPERABLE status is justified in Reference 8.~~

K.1.1, K.1.2

~~K.1~~, K.2.1 and K.2.2

INSERT K

~~Condition K applies to:~~

- ~~RWST Level - Low Low Conincident with Safety Injection, and~~
- ~~RWST Level - Low Low Coincident with Safety Injection and Coincident with Containment Sump Level - High.~~

(continued)

(B)

INSERT K

K.1.1, K.1.2

~~K.1~~ K.2.1 and K.2.2

Condition K applies to the Residual Heat Removal Pump Trip on RWST Level - Low. ~~which trips both RHR pumps.~~ Restoring the channel to OPERABLE status or placing the inoperable channel in the ~~bypass~~ cut-out condition within 6 hours is sufficient to ensure that the Function remains OPERABLE, and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed low). ~~Placing the out-of-service channel in bypass will generate a high level signal on that channel, which will ensure that under no circumstances can a failure of an additional channel low prevent the RHR pumps from starting as the result of an SI signal~~ cut-out removes that channel from the trip logic, similar to a bypass function. This provides a two-out-of-two trip logic. The 6 hour Completion Time is justified in Reference 8. If the channel cannot be placed in the ~~bypass~~ cut-out condition within 6 hours, and returned to an OPERABLE status within 72 hours, the unit must ~~immediately enter LCO 3.0.3~~ be brought to MODE 3 within the following 6 hours and MODE 5 within the next 30 hours. The 72 hour Allowed Outage Time (AOT) is the same AOT that is allowed for one inoperable RHR pump. This comparison is reasonable because the possible consequences of losing a second level channel can~~, in the worst case,~~ be no more severe than the loss of ~~one~~ a second RHR pump, and the probability of losing the level channel is ~~even~~ lower than that of losing an RHR pump. The allowed Completion Times for shutdown are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the unit does not have any analyzed transients or conditions that require the explicit use of the pump trip function noted above.

(B)

BASES

ACTIONS

K.1, K.2.1 and K.2.2 (continued)

Delete

RWST Level - Low, which trips both RHR pumps ~~Low Low Coincident With SI and Coincident With Containment Sump Level - High provides actuation of switchover to the containment sump. Note that this Function requires the bistables to energize to perform their required action. The failure of up to two channels will not prevent the operation of this Function. However, placing a failed channel in the tripped condition could result in a premature switchover to the sump, prior to the injection of the minimum volume from the RWST. Placing the inoperable channel in bypass results in a two-out-of three logic configuration, which satisfies the requirement to allow another failure without disabling actuation of the switchover when required~~. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within 6 hours is sufficient to ensure that the Function remains OPERABLE, and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed ~~high~~ low). Placing the out-of-service channel in bypass will generate a high level signal on that channel, which will ensure that under no circumstances can a failure of an additional channel low prevent the RHR pumps from starting as the result of an SI signal. The 6 hour Completion Time is justified in Reference 8. If the channel cannot be ~~returned to OPERABLE status or~~ placed in the bypass condition within 6 hours, and returned to an OPERABLE status within 72 hours, the unit must ~~be brought to MODE 3 within the following 6 hours and MODE 5 within the next 30 hours~~immediatey enter LCO 3.0.3. The 72 hour Allowed Outage Time(AOT) is the same AOT that is allowed for one inoperable RHR pump. This comparison is reasonable because the possible consequences of losing a second level channel can, in the worst case, be no more severe than the loss of one RHR pump, and the probability of losing the level channel is even lower than that of losing an RHR pump. The allowed Completion Times for shutdown are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the unit does not have any analyzed transients or conditions that require the explicit use of the ~~protection~~pump trip function~~s~~ noted above.

The Required Actions are modified by a Note that allows placing a ~~second~~ channel in the bypass condition for up to 4 hours for surveillance testing. The total of 12 hours to reach MODE 3 and 4 hours for a ~~second~~ channel to be bypassed is acceptable based on the results of Reference 8.

(continued)

C

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.8

SR 3.3.2.8 is the performance of a TADOT (except AFW; see SR 3.3.2.13). This test is a check of the Manual Actuation Functions ~~and AFW pump start on trip of all MFW pumps~~. It is performed every ~~[18]~~ 24 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.). The Frequency is adequate, based on industry operating experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a CHANNEL CALIBRATION.

A CHANNEL CALIBRATION is performed every ~~[~~18~~]~~ months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. ~~The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.~~

The Frequency of ~~[~~18~~]~~ months is based on the assumption of an ~~[~~18~~]~~ month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.2.10

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the

(continued)

(B)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.10 (continued)

accident analysis. Response Time testing acceptance criteria are included in the ~~Technical Requirements Manual, Section 15 (Ref. 9)~~ FSAR and SR 3.3.2.10 is only applicable to those functions with a specified limit. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

INSERT TSTF-111

ESF RESPONSE TIME tests are conducted on an ~~[18]~~ month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every ~~[18]~~ months. The ~~[18]~~ month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching ~~[1000]~~ 650 psig in the SGs.

SR 3.3.2.11

SR 3.3.2.11 is the performance of a TADOT as described in SR 3.3.2.8, except that it is performed for the P-4 Reactor

Each verification shall include at least one train such that both trains are verified at least once per 48 months.

(B)

Insert # TSTF-111 (Insert. pp. B 3.3-128)

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," dated January 1996, provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

The allocations for sensor response times must be verified prior to placing the component in initial operational service and re-verified following maintenance that my adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

(B)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.11 (continued)

Trip Interlock~~, and the Frequency is once per RTB cycle~~. ~~This~~ The ~~18~~ 24 month Frequency is based on operating experience. ~~demonstrating that undetected failure of the P-4 interlock sometimes occurs when the RTB is cycled.~~

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

Insert B

REFERENCES

1. FSAR, Chapter 6.
2. FSAR, Chapter 7.
3. FSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. ~~RTS/ESFAS Setpoint Methodology Study.~~ WCAP-11082, Rev. ~~2~~ 5, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Stations - Eagle 21 Version," May 1993
7. ~~NUREG-1218, April 1988.~~ WCAP-13900, "Extension of Slave Relay Surveillance Test intervals", April 1994
8. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
9. ~~Technical Requirements Manual, Section 15, "Response Times."None.~~ WCAP-13878, "reliability of Potter & Brumfield MDR Relays", June 1994.
10. WCAP-14117, "Reliability Assessment of Potter and Brumfield MDR Series Relays."
11. **WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.**
12. WCAP-11082, Rev.5, "Westinghouse Setpoint Methodology for Protection Systems, Diablo Canyon Units 1 and 2, 24 Month Fuel Cycle Evaluation," January 1997

B

Insert B

SR 3.3.2.12

SR 3.3.2.12 is the performance of an ACTUATION LOGIC TEST as described in TS 1.0 "Definitions." This SR is applied to the RHR Pump Trip on RWST Level-Low actuation logic and relays which are not processed through the SSPS. This test is performed every refueling outage. The frequency is adequate based on site and industry operating experience, considering equipment reliability and history data.

SR 3.3.2.13

SR 3.3.2.~~8~~13 is the performance of a TADOT for. This test is a check of the Manual Actuation Function~~s~~ ~~and~~ AFW ~~pump start on trip of all MFW pumps~~. It is performed every 9~~[18]~~ months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.). The Frequency is adequate, based on industry operating experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT for manual initiation Functions. The manual initiation Functions have no associated setpoints.

E

B 3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified ~~by unit specific documents~~in the FSAR section 7.5 (Ref. 1) ~~addressing~~based upon the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO include two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and/Category I variables. or

Type A variables are included in this LCO because they provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for DBAs. ~~Because the list of Type A variables differs widely between units, Table 3.3.3-1 in the accompanying LCO contains no examples of Type A variables, except for those that may also be Category I variables.~~

Category I variables are the key variables deemed risk significant because they are needed to:

(continued)

E

BASES

LCO

5. Reactor Coolant System Pressure (Wide Range) (continued)

~~In some units,~~ RCS pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture (SGTR) or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication. Furthermore, RCS pressure is one factor that may be used in decisions to terminate RCP operation.

6. Reactor Vessel Water Level Indication System (RVLIS)

~~Reactor Vessel Water Level~~ RVLIS is provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy.

The ~~Reactor Vessel Water Level Monitoring System~~RVLIS provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory.

Reactor Cavity strike-out remove strike-out

7. b. ~~a.~~ Containment ~~Wide Range~~ Sump Water Level ~~(Wide Range)~~ and
a. ~~b.~~ Containment Recirculation Sump Level (Narrow Range)

Reactor Cavity

Containment (Wide Range) Sump Water Level is provided for verification and long term surveillance of RCS integrity.

more to be last ¶

~~Containment Sump Water Level is used to determine:~~

- ~~containment sump level accident diagnosis;~~
- ~~when to begin the recirculation procedure; and~~
- ~~whether to terminate SI, if still in progress.~~

Containment Recirculation Sump Level (Narrow Range) is used to verify that sufficient water is contained in the recirculation sump to allow operation of the Residual Heat Removal Pumps with the suction aligned to the containment recirculation sump. The required Regulatory Guide 1.97 recorder for this Function is part of this instrument channel.

Containment

The Reactor Cavity Sump level instrumentation encompasses the range of the Containment Recirculation Sump and can be

(continued)

(E)

BASES

used to determine the appopriate time for swap-over of the RHR pumps from RWST to the Contanment Recirculation Sump if required.

LCO (continued)

8. a. Containment Pressure (Wide Range) and b. Containment Pressure (Normal Range)

Containment Pressure ~~(Wide Range)~~ is provided for verification of RCS and containment OPERABILITY.

Containment pressure is used to verify closure of main steam isolation valves (MSIVs) during a main steam line break inside containment, and containment spray Phase B isolation when ~~High-3~~ high-high containment pressure is reached.

Both instruments are required to cover the Regulatory Guide 1.97 range requirements.

9. Containment Isolation Valve Position

CIV Position is provided for verification of Containment OPERABILITY, and Phase A and Phase B isolation, and containment ventilation system isolation.

When used to verify Phase A and Phase B isolation, the important information is the isolation status of the containment penetrations. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two active valves. For containment penetrations with only one active CIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve, as applicable, and prior knowledge of a passive valve, or via system boundary status. If a normally active CIV is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE. This Function is on a per valve basis and ACTION A. is entered seperately for each inoperable valve indication. Note (a) to the Required Channels states that the Function is not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(continued)

(E)

BASES

LCO (continued)

10. Containment Area Radiation (High Range)

Containment Area Radiation is provided to monitor for the potential of significant radiation releases ~~and to provide release assessment~~ for use by operators in determining the need to invoke site emergency plans. Containment radiation level is used to determine if a high energy line break (HELB) containing radioactive fluid has occurred, and whether the event is inside or outside of containment.

11. ~~Containment~~ Hydrogen ~~Concentration~~~~Monitors~~ remove strike out

~~Containment~~ Hydrogen ~~Monitors are~~ ~~Concentration~~ monitoring is provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions and is used to determine whether or not hydrogen recombiners should be started.

12. Pressurizer Level

Pressurizer Level is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition.

13. a. Steam Generator Water Level (Wide Range) and b. Steam Generator Level (Narrow Range)

SG Water Level is provided to monitor operation of decay heat removal via the SGs. ~~The Category I indication of SG level is the extended startup range level instrumentation.~~ The ~~extended startup~~ wide range level covers a span of 12 inches ~~≥ 6 inches to ≤ 394 582 inches~~ above the lower tubesheet to the steam generator separators. The measured differential pressure is displayed in ~~inches of water at 68°F.~~ percent level (cold calibration).

~~Temperature compensation of this indication is performed manually by the operator. Redundant monitoring capability is provided by two trains of instrumentation. The uncompensated level signal is~~

(continued)

(E)

BASES

LCO

19. Auxiliary Feedwater Flow (continued)

~~At some units,~~ AFW flow is a Type A variable because operator action is required to throttle flow during an SLB accident to prevent the AFW pumps from operating in runout conditions. AFW flow is also used by the operator to verify that the AFW System is delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level (Narrow Range) during normal SG inventory conditions.

20. (new) Refueling Water Storage Tank (RWST) Water Level

RWST water level is used to verify the water source availability to the emergency core cooling system (ECCS) and Containment Spray Systems. It may also provide an indication of time for initiating cold leg recirculation from the sump following a LOCA. The RWST level ~~signal~~ channel additionally trips the Residual Heat Removal Pumps ~~at 33%~~ in preparation for transfer to cold leg recirculation. add strike-out

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3 except for the ~~Containment~~ ~~H~~ydrogen ~~Concentration~~ monitors ~~that is~~ are only required to be OPERABLE in MODES 1 and 2. ~~These variables~~ are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, and in MODE ~~3~~ for the ~~Containment~~ Hydrogen ~~Concentration~~ monitor, unit conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

Note 1 has been added in the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident using alternate instruments and methods, and the low probability of an event requiring these instruments.

Note 2 has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1.

(continued)

(E)

B 3.3 INSTRUMENTATION

B 3.3.4 Remote Shutdown System

BASES

BACKGROUND

The Remote Shutdown System provides the control room operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible. A safe shutdown condition is defined as MODE 3. With the unit in MODE 3, the Auxiliary Feedwater (AFW) System and the steam generator (SG) safety valves ~~or the SG atmospheric dump valves (ADVs)~~ can be used to remove core decay heat and meet all safety requirements. The long term supply of water for the AFW System ~~and the ability to borate the Reactor Coolant System (RCS)~~ allows extended operation in MODE 3 until such time that either control is transferred back to the Control Room or a cooldown is initiated from outside the control room.

If the control room becomes inaccessible, the operators can establish control at the remote shutdown panel (hot shutdown panel), and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located at the ~~remote~~ hot shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. The unit automatically reaches MODE 3 following a unit shutdown and can be maintained safely in MODE 3 for an extended period of time.

The OPERABILITY of the following remote shutdown control and instrumentation functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible.

Insert A: →

APPLICABLE SAFETY ANALYSES

The Remote Shutdown ~~System is required to~~ Instrumentation Functions and the hot shutdown panel controls provide~~s~~ equipment at appropriate locations outside the control room with a capability to promptly shut down and maintain the unit in a safe condition in MODE 3.

The criteria governing the design and specific system requirements of the Remote Shutdown ~~System~~ Instrumentation Functions and controls are located in 10 CFR 50, Appendix A, GDC 19 (Ref. 1).

(continued)

INSERT A

E

INSTRUMENT/CONTROL FUNCTION	READOUT/CONTROL LOCATION	REQUIRED NUMBER OF CHANNELS
1. Reactor Trip Breaker Indication	Reactor Trip Breaker	1/trip breaker
2. Pressurizer Pressure	Hot Shutdown Panel	1
3. Pressurizer Level	Hot Shutdown Panel	1
4. Steam Generator Pressure	Hot Shutdown Panel	1/stm. gen.
5. Steam Generator Wide Range Water Level OR Auxiliary Feedwater Flow	Hot Shutdown Panel	1/stm. gen.
6. Condensate Storage Tank Water Level	Hot Shutdown Panel	1
~~7. Auxiliary Feedwater Flow~~	~~Hot Shutdown Panel~~	~~1/stm. gen.~~
7 ~~8~~. Charging Flow	Hot Shutdown Panel	1
8 ~~9~~. RCS Loop 1 Temperature Indication	Dedicated Shutdown Panel	Hot and Cold Leg Temperature Indication
9 ~~10~~. Auxiliary Feedwater Flow Control - AFW Pump, and Associated Valves - Transfer Switches	 Hot Shutdown Panel 4kV Switchgear	 any 2 of 3 AFW pumps
10 ~~11~~. Charging Flow Control - Centrifugal Charging Pump - Transfer Switch	 Hot Shutdown Panel 4kV Switchgear	 2 of 2 pumps
11 ~~12~~. Component Cooling Water Control - Component Cooling Water Pump - Transfer Switch	 Hot Shutdown Panel 4kV Switchgear	 any 2 of 3 CCW pumps
12 ~~13~~. Auxiliary Saltwater Control - Auxiliary Saltwater Pump - Transfer Switch	 Hot Shutdown Panel 4kV Switchgear	 2 of 2 pumps
13 ~~14~~. Emergency Diesel Generator Control - EDG Start	 EDG Local Control Panel	 3 of 3 EDGs

E

BASES

APPLICABLE SAFETY ANALYSES (continued)

are

The Remote Shutdown ~~System~~ Instrumentation Functions and the hot shutdown panel controls ~~is~~ considered an important contributor to the reduction of unit risk to accidents and as such it has been retained in the Technical Specifications as indicated ~~in the NRC Policy Statement~~by Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The Remote Shutdown ~~System~~ Instrumentation Functions and the hot shutdown panel controls LCO provides the OPERABILITY requirements of the instrumentation and controls necessary to place and maintain the unit in MODE 3 from a location other than the control room. The instrumentation and controls ~~typically~~ required are listed in Table 3.3.4-1 in the accompanying LCO.

~~Reviewer's Note: For channels that fulfill GDC 19 requirements, the number of OPERABLE channels required depends upon the unit licensing basis as described in the NRC unit specific Safety Evaluation Report (SER). Generally, two divisions are required OPERABLE. However, only one channel per a given Function is required if the unit has justified such a design, and NRC's SER accepted the justification.~~

The controls, instrumentation, and transfer switches are required for:

- ~~Core reactivity control (initial and long term)~~ Reactor trip indication;
- RCS pressure control;
- Decay heat removal via the AFW System and the SG safety valves ~~or SG ADVs~~;
- RCS inventory control via charging flow; and
- Safety support systems for the above Functions, including ~~service water~~ auxiliary saltwater, component cooling water, and ~~onsite power, including the~~ diesel generators.

A Function of a Remote Shutdown System is OPERABLE if all required instrument and control channels ~~needed to support the Remote Shutdown System Function~~ for that function listed in Table 3.3.4-1 are OPERABLE. In some cases, Table 3.3.4-1 may indicate that the required information or control capability is available from several alternate sources. In these cases, the Function is OPERABLE as long

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.3 (continued)

The Frequency of [18] months is based upon operating experience and consistency with the typical industry refueling cycle.

~~SR 3.3.4.4~~

~~SR 3.3.4.4 is the performance of a TADOT every 18 months. This test should verify the OPERABILITY of the reactor trip breakers (RTBs) open and closed indication on the remote shutdown panel, by actuating the RTBs. The Frequency is based upon operating experience and consistency with the typical industry refueling outage.~~

NOTE: A surveillance of the reactor trip breaker OPERABILITY is not required as part of the SURVEILLANCE REQUIREMENT since a TRIP ACTUATING DEVICE OPERATIONAL TEST of the reactor trip breakers is performed as part of the SURVEILLANCE REQUIREMENT for TS 3.3.1

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19 (associated with 1967 GDC 11 per FSAR Appendix 3.1A)

B

BASES

Trip Setpoints and Allowable Values

The Trip Setpoints used in the relays are based on the analytical limits presented in FSAR, Chapter 15 (Ref. 2). The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account.

The actual nominal Trip Setpoint entered into the relays is normally still more conservative than that required by the Allowable Value. If the measured setpoint does not exceed the Allowable Value, the undervoltage relay is considered OPERABLE. If the measured time delay does not exceed the Allowable Value, the timer is considered OPERABLE.

Setpoints adjusted in accordance with the Allowable Value ensure that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the accident and that the equipment functions as designed.

Allowable Values and/or Trip Setpoints are specified for each Function in the LCO. ~~Nominal Trip Setpoints are also specified in the unit specific setpoint calculations~~. The nominal setpoints are selected to ensure that the setpoint measured by the surveillance procedure does not exceed the Allowable Value if the undervoltage relay is performing as required. If the measured setpoint does not exceed the Allowable Value, the undervoltage relay is considered OPERABLE. Operation with a Trip Setpoint less conservative than the nominal Trip Setpoint, but within the Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculation. Each Allowable Value and/or Trip Setpoint specified is more conservative than the analytical limit assumed in the transient and accident analyses in order to account for instrument uncertainties appropriate to the trip function. These uncertainties are defined in ~~the "Unit Specific RTS/ESFAS Setpoint Methodology Study"~~ WCAP 11082, Rev. 2 "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Stations Eagle 21 Version Q" (Ref. 3). remove strike-out

APPLICABLE SAFETY ANALYSES

The LOP DG start instrumentation is required for the Engineered Safety Features (ESF) Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS).

Accident analyses credit the loading of the DG based on the loss of offsite power during a loss of coolant accident (LOCA). The actual DG start has historically been associated with the ESFAS actuation. The DG loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. The analyses assume a non-mechanistic DG loading, which does not explicitly account for each individual component of loss of power detection and subsequent actions.

(continued)

(B) & (E)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LOP DG start instrumentation channels satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2)(ii)

LCO

for — and their three corresponding timers

The LCO for LOP DG start instrumentation requires that ~~[three]~~ ~~one~~ two channels per bus ~~of~~ ~~both the~~ loss of voltage and two channels per bus for initiation of load shed and two channels per bus of degraded voltage with one timer per bus for DG start and initiation of load shed Functions shall be OPERABLE in MODES 1, 2, 3, and 4 when the LOP DG start instrumentation supports safety systems associated with the ESFAS. In MODES 5 and 6, the ~~[three]~~ channels must be OPERABLE whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed. Loss of the LOP DG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power the DG powers the motor driven auxiliary feedwater pumps. Failure of these pumps to start would leave only one turbine driven pump, as well as an increased potential for a loss of decay heat removal through the secondary system.

APPLICABILITY

The LOP DG Start Instrumentation Functions are required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE so that it can perform its function on an LOP or degraded power to the vital bus.

ACTIONS

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the channel is found inoperable, then the function that channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected.

Because the required channels are specified on a per bus basis, the Condition may be entered separately for each bus as appropriate.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of

(continued)

A

BASES

ACTIONS (continued)

~~C.1~~

~~Condition C applies to each of the LOP DG start Functions when the Required Action and associated Completion Time for Condition A or B are not met.~~

In these circumstances the Conditions specified in LCO 3.8.1, "AC Sources – Operating," or LCO 3.8.2, "AC Sources – Shutdown," for the DG made inoperable by failure of the LOP ~~DG start~~ instrumentation are required to be entered immediately. The actions of those LCOs provide for adequate compensatory actions to assure unit safety.

A Note is added to allow bypassing an inoperable channel for up to 2 hours for surveillance testing. This allowance is made where bypassing the channel does not cause an actuation and where at least one other channel is monitoring that parameter.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1 NOT USED

~~SR 3.3.5.1~~

~~Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.~~

~~Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.~~

~~The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~

(continued)

(E)

BASES

BACKGROUND (continued)

~~purge~~ ventilation isolation, which closes ~~both inner and outer~~ the containment ventilation isolation valves ~~in the Mini Purge System and the Shutdown Purge System~~. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event, within approximately 60 seconds. The isolation of the ~~purge~~ containment ventilation valves has not been analyzed mechanistically in the dose calculations, although its ~~rapid~~ isolation, using a conservative isolation time, is assumed. The containment purge and exhaust ventilation isolation radiation monitors act as backup to the SI signal to ensure closing of the purge and exhaust containment ventilation isolation valves following a LOCA. They are also the primary means for automatically isolating containment in the event of a fuel handling accident or any other source within containment during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The containment purge and exhaust ventilation isolation instrumentation satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Purge and Exhaust Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation
NOT USED

~~The LCO requires two channels OPERABLE. The operator can initiate Containment Purge Isolation at any time by using either of two switches in the control room. Either switch actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.~~

~~The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.~~

~~Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet.~~

(continued)

(E)

BASES (continued)

APPLICABILITY

The ~~Manual Initiation,~~ Automatic Actuation Logic and Actuation Relays, Containment Isolation – Phase A, and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4, and during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the containment purge and exhaust ventilation isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6 without fuel handling in progress, the containment purge and exhaust ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a ~~COT~~CFT and/or Channel Calibration, when the process instrumentation is set up for adjustment to bring it within specification. Drift can also be observed during a Channel Check or CFT and if observed would prompt action to correct the discrepancy. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered. *

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one containment ~~purge~~ ventilation isolation radiation monitor channel. ~~Since the four containment radiation monitors measure different parameters,~~

(continued)

(E)

B 3.3 INSTRUMENTATION

B 3.3.7 Control Room ~~Emergency Filtration~~ Ventilation System (~~CREFS~~ CRVS) Actuation Instrumentation

BASES

BACKGROUND

The ~~CREFS~~ CRVS provides an enclosed control room environment from which ~~the~~both units can be operated following an uncontrolled release of radioactivity. ~~During normal operation, the Auxiliary Building Ventilation System provides control room ventilation.~~ Upon receipt of an actuation signal, the ~~CREFS~~ CRVS shifts from normal operation and initiates filtered ventilation and pressurization of the control room. This system is described in the Bases for LCO 3.7.10, "Control Room ~~Emergency Filtration~~ Ventilation System~~.~~" and is common to both units.

The actuation instrumentation consists of redundant radiation monitors in the air intakes ~~and~~ to the control room areas. There are two detectors in each of the two normal control room air intakes. However, since they take suction form a common area, the North and South sides of the mechanical equipment room, only two detectors are required to provide protection against a single failure. A Phase "A" containment isolation signal or a high radiation signal from ~~any~~either of these required detectors in the normal intake will initiate ~~both trains of the CREFS~~ CRVS pressurization from the pressurization intake with the lowest radiation level (each pressurization intake, one on the North end of the turbine building and one on the South, has two radiation monitors). The control room operator can also initiate ~~CREFS~~ CRVS pressurization ~~trains~~ by manual switches in the control room. ~~The CREFS is also actuated by a safety injection (SI) signal. The SI Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."~~

The CRVS has two additional manually selected operating modes; smoke removal and recirculation. Neither of these modes are required for the CRVS to be OPERABLE, but they are useful for certain non-DBA circumstances.

APPLICABLE SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

The ~~CREFS~~ CRVS acts to terminate the supply of unfiltered outside air to the control room, initiate filtration, and pressurize the control room. These actions are necessary to ensure the control room is kept habitable for the operators stationed there during accident recovery and post accident operations by minimizing the radiation exposure of control room personnel.

(continued)

(E)

BASES

In MODES 1, 2, 3, and 4, the radiation monitor actuation of the ~~CREFS~~ CRVS is a backup for the ~~SI~~ Phase A signal actuation. This ensures initiation of the ~~CREFS~~ CRVS during a loss of coolant an e accident or steam generator tube rupture involving a release of radioactive materials.

The radiation monitor actuation of the CREFS in MODES 5 and 6, during movement of irradiated fuel assemblies ~~[~~, and

(continued)

D

BASES

APPLICABLE SAFETY ANALYSES (continued)

CORE ALTERATIONS~~]~~. is the primary means to ensure control room habitability in the event of a fuel handling or waste gas decay tank rupture accident. The ~~CREFS~~ CRVS pressurization system actuation instrumentation satisfies Criterion 3 ~~of the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

remove strike-out

LCO

The LCO requirements ensure that instrumentation necessary to initiate the ~~CREFS~~ CRVS pressurization system is OPERABLE.

1. Manual Initiation

remove strike-out

The LCO requires two channels OPERABLE. The operator can initiate the ~~CREFS~~ CRVS pressurization mode at any time by using ~~either of~~ two switches in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

~~Each channel consists of one push button and the interconnecting wiring to the actuation logic cabinet.~~

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Actuation Logic and Relays OPERABLE to ensure that no single random failure can prevent automatic actuation of the pressurization system.

~~Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b., SI, in LCO 3.3.2. The applicable MODES and specified conditions for the CREFS portion of these functions are different and less restrictive than those specified for their SI roles. If one or more of the SI functions becomes inoperable in such a manner that only the CREFS function is affected, the Conditions applicable to their SI function need not be entered. The less~~

(continued)

(A)

(B)

BASES

ACTIONS (continued)

B.1.1 and B.1.2 ~~and B.2~~

Condition B applies to the failure of two ~~CREFS~~ CRVS actuation trains, two radiation monitor channels, or two manual channels. The first Required Action is to place one ~~CREFS~~ CRVS train in the ~~emergency [radiation protection]~~ pressurization mode of operation immediately. This accomplishes the actuation instrumentation Function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required Actions of LCO 3.7.10 must also be entered for the ~~CREFS~~ CRVS train made inoperable by the inoperable actuation instrumentation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.10.

~~Alternatively,~~ Both trains cannot ~~may~~ be placed in the ~~emergency [radiation protection]~~ pressurization mode. ~~This ensures the CREFS function is performed even in the presence of a single failure.~~

Since the design of the system is such that operation of two pressurization fans would overpressurize the supply ducting to the filters.

~~The Required Action for Condition B is modified by a Note that requires placing one CREFS train in the toxic gas protection mode instead of the [radiation protection] mode of operation if the automatic transfer to toxic gas protection mode is inoperable. This ensures the CREFS train is placed in the most conservative mode of operation relative to the OPERABILITY of the associated actuation instrumentation.~~

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

D

B 3.3 INSTRUMENTATION

B 3.3.8 Fuel Handling Building ~~Air Cleanup~~Ventilation System (~~FBACS~~FHBVS) Actuation Instrumentation

BASES

BACKGROUND

The ~~FBACS~~FHBVS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident ~~or a loss of coolant accident (LOCA)~~ are filtered and adsorbed prior to exhausting to the environment. The system is described in the Bases for LCO 3.7.13, "Fuel Handling Building ~~Air Cleanup~~Ventilation System." The system initiates filtered ventilation of the fuel building automatically following receipt of a high radiation signal ~~(gaseous or particulate) or a safety injection (SI) signal~~ from the Spent Fuel Pool Monitor or from the New Fuel Storage Vault Monitor (or from gaseous monitors 45 A/B when installed). Initiation may also be performed manually as needed from the main control room or fuel handling building.

High ~~gaseous and particulate~~ radiation, ~~each monitored by~~ from either of the two monitors, provides ~~FBACS~~FHBVS initiation. ~~Each FBACS train is initiated by high radiation detected by a channel dedicated to that train. There are a total of two channels, one for each train. Each channel contains a gaseous and particulate monitor.~~ High ~~radiation detected by any monitor~~ or an SI signal from the Engineered Safety Features Actuation System (ESFAS) initiates fuel building isolation and starts the FBACS. These actions function to prevent exfiltration of contaminated air by initiating filtered ventilation, which imposes a negative pressure on the fuel building. ~~Since the radiation monitors include an air sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.~~

should be strike-out

APPLICABLE SAFETY ANALYSES

The ~~FBACS~~FHBVS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident ~~or a LOCA~~ are filtered and adsorbed prior to being exhausted to the environment. This action reduces the radioactive content in the fuel building exhaust following a ~~LOCA or~~ fuel handling accident so that offsite doses remain within the limits specified in 10 CFR 100 (Ref. 1).

The ~~FBACS~~FHBVS actuation instrumentation satisfies Criterion 3 of ~~the NRC Policy Statement~~10 CFR 50.36(c)(2)(ii).

(continued)

(B)

BASES

LCO ~~3~~ 2. Fuel Building Radiation (continued)

~~For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, filter motor operation, detector OPERABILITY, if these supporting features are necessary for actuation to occur under the conditions assumed by the safety analyses.~~

Only the Trip Setpoint is specified for each ~~FBACS~~FHBVS Function in the LCO. ~~The Trip Setpoint limits account for instrument uncertainties, which are defined in the Unit Specific Setpoint Calibration Procedure (Ref. 2).~~ add strike-out

APPLICABILITY The manual ~~FBACS~~ FBVS initiation must be OPERABLE ~~in MODES [1, 2, 3, and 4] and~~ when moving irradiated fuel assemblies in the fuel building, to ensure the ~~FBACS~~FHBVS operates to remove fission products associated with ~~leakage after a LOCA or~~ a fuel handling accident. ~~The automatic FBACS actuation instrumentation is also required in MODES [1, 2, 3, and 4] to remove fission products caused by post LOCA Emergency Core Cooling Systems leakage.~~

High radiation initiation of the ~~FBACS~~FHBVS must be OPERABLE in any MODE during movement of irradiated fuel assemblies in the fuel building to ensure automatic initiation of the ~~FBACS~~ FBVS when the potential for a fuel handling accident exists.

While in MODES 5 and 6 without fuel handling in progress, the ~~FBACS~~FHBVS instrumentation need not be OPERABLE since a fuel handling accident cannot occur.

ACTIONS The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a ~~COT~~CFT and/or Channel Calibration, when the process instrumentation is set up for adjustment to bring it within

(E)

(continued)

8

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 (continued)

channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.8.2

A ~~COT~~ CFT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the FBACS actuation. ~~The setpoints shall be left consistent with the unit specific calibration procedure tolerance.~~ The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.8.3 Not Used

~~SR 3.3.8.3~~

~~SR 3.3.8.3 is the performance of an ACTUATION LOGIC TEST. The actuation logic is tested every 31 days on a STAGGERED TEST BASIS. All possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency is based on the known~~

(continued)

BASES

~~SURVEILLANCE~~ REQUIREMENTS

~~SR 3.3.8.3 (continued)~~

~~reliability of the relays and controls and the multichannel redundancy available, and has been shown to be acceptable through operating experience.~~

SR 3.3.8.4

SR 3.3.8.4 is the performance of a TADOT. This test is a check of the manual actuation functions and is performed every ~~[18]~~ months. Each manual actuation function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Frequency is based on operating experience and is consistent with the typical industry refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.8.5

A CHANNEL CALIBRATION is performed every ~~[18]~~ months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.

~~2. Unit Specific Setpoint Calibration Procedure.~~

(E)

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1431

NUREG-1431 Section 3.3

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

CHANGE NUMBER	JUSTIFICATION
3.3-01	This trip function or design feature is not included in the plant design or it is not credited and has no safety function.
3.3-02	For the Reactor Trip on Turbine Trip function based on turbine stop valve position, 4 of 4 channels are required to close to less than 1% open in order to generate the reactor trip signal. Thus, it is acceptable to have more than one Turbine Stop Valve Closure - reactor trip function channel inoperable and placed in trip per current TS Table 3.3-1, Functional Unit [17.b], ACTION Statement [7]. In addition, the 4 hour bypass note applies only to the [Low Auto Stop Oil Pressure] channels. ITS 3.3.1 Condition P has been revised.
3.3-03	This change to ITS 3.3.1 Condition R is consistent with the current licensing basis. A 4-hour AOT for SSPS logic surveillance testing has little usefulness if the RTBs cannot be bypassed for the duration of that testing. RTB surveillance testing retains the current 2-hour AOT.
3.3-04	Not applicable to Diablo Canyon Power Plant (DCPP). See Conversion Comparision Table (Enclosure 6B).
3.3-05	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-06	Retains CTS power requirement of 75% RTP in the ITS SR 3.3.1.6 Note concerning when the incore/excore calibration is performed. The ISTS proposal would require unnecessary delays in the post-refuel power ascension. As per the current TS 4.0.4 exception, it is acceptable to go above 75% RTP during power ascension provided the calibration is performed within 24 hours of exceeding 75% RTP. ~~The Note is further revised to permit achieving equilibrium conditions (per CTS 4.2.2.2.d.1) prior to performing the required surveillance.~~
3.3-07	Note 3 is added to ITS SR 3.3.1.11 to be consistent with the CTS Table 4.3-1 Note [5]. This ensures that this exception for power and intermediate range detector plateau voltage verification, as discussed in the ITS BASES for SR 3.3.1.11, is included in the Technical Specifications rather than being only found in the BASES. The note replaces the exception to LCO 3.0.4 in the current TS.
3.3-08	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-09	The addition of footnote [(m)] to ITS Table 3.3.1-1 for Function 10 clarifies the low flow setpoint relationship to the quantity identified as Minimum Measured Flow, consistent with the current TS.

(B) & (E)

CHANGE NUMBER	JUSTIFICATION
3.3-20	This change adds note 2 on [Containment Radiation Level (High Range)] calibration in ITS SR [3.3.3.2] to be consistent with current TS Table [4.3-7 Note (2)]. This note is acceptable as it reflects the unique calibration requirements of these high range radiation monitors as defined in the current TS.
3.3-21	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-22	Consistent with CTS [3.3.3.5], [RCP breaker indication is excluded from CHANNEL CHECKS and reactor trip breaker and RCP breaker indications are excluded from] CHANNEL CALIBRATIONS in ITS SR 3.3.4.3 since these SRs have no meaning for [~~these~~ this] functions.
3.3-23	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-24	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-25	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-26	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-27	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-28	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-29	~~[Not used]~~ insert-next page
3.3-30	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B). delete
3.3-31	The current TS require the response times associated with the [~~undervoltage and degraded voltage diesel generator start functions and the~~] containment [purge and exhaust] isolation functions to be verified against the specific response time values contained in the [FSAR]. The ITS is revised to match the current TS and the response time values are [moved to the FSAR per CN 01-35-LG]. As is done with the Reactor Trip System and the ESFAS instrumentation, this method is an appropriate way to control response times. [~~SR 3.3.5.4 and~~ SR 3.3.6.8] are added to require the response time verifications.
3.3-32	Improved TS [3.3.6 ACTION A is modified by a Note and] Table 3.3.6-1 is changed to be consistent with current TS [3.3.2 Functional Unit 3.c and current TS 3.9.9]. Subfunctions [b, c and d] of Containment Radiation are stricken since only the gaseous [ ] channel provides the actuation function [and the bracketed setpoint is changed to reflect plant-specific requirements]. [The number of gaseous monitors required for CORE ALTERATIONS or during movement of irradiated fuel has been revised to one (either RM 44A or B) per the CTS].
3.3-33	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-34	This change adds an LCO 3.0.3 exception Note 1 to ITS 3.3.8 to reflect industry Traveler TSTF-36, Rev. 2.
3.3-35	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-36	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).

(B)

3.3-29 INSERT :

Functional unit 7 is revised per the DCPP current plant design to incorporate the residual heat removal (RHR) pump trip from low refueling water storage tank (RWST) level. Action K is revised and a new SR 3.3.2.12 is added. LAR 97-10 was submitted July 30, 1997 to incorporate these changes into the CTS.

CHANGE NUMBER	JUSTIFICATION
3.3-51	ITS ACTION B.2 of LCO 3.3.7 is deleted, since DCPP cannot operate with both pressurization systems running at the same time. The design of the system is such that operation of two pressurization fans would over pressurize the supply ducting to the filters.
3.3-52	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-53	The REQUIRED CHANNELS description for Functions 2.a and 3.b.(1), of ITS Table 3.3.2-1, are revised per the CTS to note that only two switches (one per train) exist and that both must be moved coincident for manual initiation.
3.3-54	Function 18.b (P-7) of ITS Table 3.3.1-1 is clarified. COTs and Channel Calibrations apply to the P-10 and P-13 inputs, not to the P-7 logic function. This change is an administrative clarification to address the relationships between these interlocks in the plant's design.
3.3-55	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-56	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-57	Not used.
3.3-58	This change adds new ITS 3.3.2 Condition [N] to reflect current TS Table 3.3-3 ACTION Statement [24] on manual AFW [and manual MSIV closure] initiation.
3.3-59	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-60	Consistent with the design and current TS, Surveillance Requirements 3.3.2.3 and 3.3.2.7 are not used by any function listed in Table 3.3.2-1 and are deleted.
3.3-61	This change revises the ITS SR 3.3.2.11 Frequency to ~~18~~ [24] months per current TS Table 4.3-2 Functional Unit [8.c], which is the ESFAS P-4 permissive. The ~~18~~ [24] month Frequency for the surveillance of the basic switch logic associated with the opening of the reactor trip breakers is the value specified in the current TS. [Deleted the Note stating that verification of set point is not required per the CTS.]
3.3-62	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-63	This change revises ITS Table 3.3.2-1 [Notes (b) and (g)] per current TS Table [3.3-3] Notes [# and ##]. This revision is a clarification to the operator that describes the circumstances under which the [Steamline Pressure Negative Rate - High, Steam Pressure-low, or Pressurizer Pressure-low functions may be or are blocked relative to the] P-11 permissive.
3.3-64	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-65	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-66	The MODE 4 requirement of the CTS is retained and added to Table 3.3.2-1 for SI actuated by Containment Pressure high-high. ITS 3.3.2 ACTIONS D and E are revised accordingly.
3.3-67	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).

E

CHANGE NUMBER	JUSTIFICATION
3.3-86	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-87	Not used.
3.3-88	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-89	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-90	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-91	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-92	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-93	ITS 3.3.1 Condition V is deleted. It is not entered from Table 3.3.1-1 nor do the Bases clarify when it would be needed, raising the concern of misinterpretation. Condition V does not replace LCO 3.0.3 requirements to assess when the plant is outside the licensing basis. There is no similar ACTION Statement in the current TS for the Reactor Trip System. This change is consistent with Traveler TSTF-135.
3.3-94	ITS 3.3.4 is revised per current TS [3.3.3.5] with regard to [remote shutdown panel] controls. [Remote shutdown panel] controls are added to the LCO, Condition A, and SR 3.3.4.2. By explicitly including the controls, the specification is clarified to be more than instrumentation. This change is acceptable because it does not change the meaning while retaining the clarity of the CTS.
3.3-95	ITS 3.3.1 Condition H, Required ACTION H.1, and the second part of Function 4 Applicability (MODE 2 below P-6) in ITS 3.3.1 are deleted since they provide no real compensatory measures. [With their deletion, there is no need to repeat the > P-6 Applicability in Conditions F and G.] In accordance with LCO 3.0.4, the intermediate range detectors must be OPERABLE prior to entering the Applicability of the retained part of Function 4 (i.e., MODE 2 above P-6). Condition H and Required ACTION H.1 ensure the same thing and, therefore, can be deleted. This change is consistent with Traveler TSTF-135.
3.3-96	[ ] Note 2 for ITS SR 3.3.1.3 is revised to replace the bracketed 15% RTP power level constraint with 50% RTP. The specified power level in ITS SR 3.3.1.3 should reflect the applicable safety analysis basis consistent with the [APPLICABILITY and] Required Actions of ITS LCO 3.2.3 (AFD) and LCO 3.2.4 (QPTR). As revised, this surveillance requirement is acceptable in that it assures the surveillance is performed after the appropriate plant conditions are attained and still provides sufficient time to perform the surveillance in a controlled manner.
3.3-97	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-98	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-99	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).

(B) & (C)

CHANGE NUMBER	JUSTIFICATION
3.3-125	ITS SR 3.3.1.11 is modified by a Note that requires verification that the time constants are adjusted to the prescribed values. The addition of this Note is consistent with SR 3.3.1.10 and is required because SR 3.3.1.11 is used for the Power Range Neutron Flux - High Positive Rate [and High Negative Rate] trip functions which have a time constant associated with their calibration.
3.3-126	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-127	The MODE 2 applicability for the undervoltage RCP start of the steam-driven AFW pump is deleted and the surveillance Frequency is revised per the DCPP CTS. Thus, the Required Actions of ACTION I are revised to include entering MODE 2 for function 6.g and MODE 3 for function 5.b, and the required surveillance is changed from SR 3.3.2.7 to SR 3.3.2.8. This anticipatory start of the steam-driven AFW pump is not credited for MODE 2 operation, only the SG low level start signal is used for MODE 2 or 3.
3.3-128	This change revises ITS Table 3.3.4-1 to be consistent with CTS 3.3.3.5.
3.3-129	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-130	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-131	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-132	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-133	This change revises ITS LCO 3.3.5 and SR 3.3.5.3 to include the DG start sequence delay timers from CTS Table 3.3-4.
3.3-134	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
3.3-135	A MODE change restriction has been added to ITS 3.3.1 Condition C per the matrix discussed in CN 1-02-LS-1 of the 3.0 package (see LS-1 NSHC in the CTS Section 3/4.0, ITS Section 3.0 package).
3.3-136	Not applicable to DCPP. See Conversion Comparision Table (Enclosure 6B).
{3.3-137	The Condition for Function 4.c is changed from Condition D to E consistent with the CTS. Plant design requires this Function to be bypassed, not tripped if inoperable.}
3.3-139	This change adds new SR 3.3.2.13 which is the performance of an 18 month TADOT. SR 3.3.2.8 is the performance of a TADOT every 24 months. As part of the DCPP 24 month fuel cycle evaluations, the AFW manual actuation function will remain at 18 months.

[illegible]

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-07	Note 3 is added to ITS SR 3.3.1.11 to be consistent with the CTS Table 4.3-1 Note [5]. This ensures that this exception, for power and intermediate range detector plateau voltage verification as discussed in the ITS BASES for SR 3.3.1.11, is included in the Technical Specifications rather than being only found in the BASES.	Yes	Yes	Yes	Yes
3.3-08	Deletes the Reviewer's Note in ITS Tables 3.3.1-1 and 3.3.2-1 and adds a Note reflecting the Allowable Value as the LSSS. Trip Setpoints are listed in the Bases.	No, retained CTS format.	Yes	Yes	Yes
3.3-09	The addition of footnote [(m)] to ITS Table 3.3.1-1 for Function 10 clarifies the low flow setpoint relationship to the quantity identified as Minimum Measured Flow, consistent with the CTS.	Yes	No, not in CTS.	No, not in CTS.	Yes, (CTS per OL Amendment No. 15 dated 4-8-86)
3.3-10	[The Overtemperature ΔT setpoint equation had a bracket in the wrong place and was corrected.] In addition, the f_1 (ΔI) penalty function was corrected and the K_2 inequality sign was changed to an equal sign.	Yes	No, see CN 3.3-38.	Yes	Yes, (CTS per OL Amendment No. 102 dated 8-21-95)
3.3-11	Added "or Rod Control System incapable of rod withdrawal," which makes Note (f) the complete antithesis of Note (b).	Yes	No, see CN 3.3-41.	No, see CN 3.3-41.	No, see CN 3.3-41.
3.3-12	Corrects typo in the inequality sign of ITS Table 3.3.2-1 Note (h).	No, see CN 3.3-105.	Yes	Yes	Yes
3.3-13	The equations for Overtemperature ΔT and Overpower ΔT are revised to be consistent with the DCPP CTS. The value of the time constant τ_6 has always been 0 seconds and the factor utilizing the time constant has not been shown as part of the equation in licensing documents since the factor value has been unity. Thus, the factors utilizing the time constant has been deleted.	Yes	No	No	No
3.3-14	Retains the monthly COT for Function 6.h of ITS Table 3.3.2-1, per CTS Table 4.3-2 Functional Unit 6.h. No TADOT is performed.	No, not in CTS.	No, not in CTS.	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-26	The Pressurizer PORV and Block Valve Controls are deleted and RCP Breaker Position and AFW Suction Pressure are added to ITS Table 3.3.4-1, consistent with the current licensing basis for compliance with GDC-19. The PORVs may be used for eventual plant cooldown; however, they are not required to attain HOT STANDBY which is the basis for the listed functions. The added functions may be used to ensure decay heat removal by the SGs in attaining HOT STANDBY.	No, not in CTS.	No, see CN 3.3-24.	Yes	Yes (per FSAR Section 7.4.3 and SER Sections 7.4.2 and 7.4.3.2)
3.3-27	This change modifies ITS SR 3.3.3.2 and SR 3.3.3.3 to allow for different surveillance frequencies for the hydrogen monitors than other PAMS components. The manufacturer for the CPSES hydrogen monitors specifies a more frequent calibration frequency than that required for the other PAMS instruments. The more frequent calibration is required to assure function operability.	No	Yes	No	No
3.3-28	Tie breaker changes per CTS Table 3.3-3, ACTION Statement 19 for Functional Unit 8.b.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes (CTS per OL Amendment No. 99 dated 4-18-95)
3.3-29	~~(Not used)~~	~~N/A~~ Yes	~~N/A~~ No	~~N/A~~ No	~~N/A~~ No
3.3-30	The portion of Condition C [(relabeled as D per CN 3.3-74)] referring to one or more functions with one or more automatic actuation trains inoperable is revised to cover BOP-ESFAS only.	No, not in CTS.	No, not in CTS.	Yes	Yes

Functional unit 7 is revised per the DCPP current plant design to incorporate the residual heat removal (RHR) pump trip from low refueling water storage tank level. Action K is revised and new SR 3.3.2.12 is added. LAR 97-10 was submitted July 30, 1997 to incorporate these changes into the CTS.

B

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-36	Revisions reflect revised BDMS setpoint in CTS.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes (CTS per OL Amendment No. 94 dated 3-7-95)
3.3-37	delete Several ITS Required Action Notes are modified to allow a channel to be placed in bypass for surveillance testing. ~~[This change is consistent with the CTS.]~~	Yes	Yes	No, not in current design or TS.	No, not in current design or TS.
3.3-38	The CPSES design uses the N-16 based overtemperature and overpower protective functions. Several changes to the setpoints, Required Actions and Surveillances of NUREG-1431 are required to maintain the current licensing basis.	No	Yes	No	No
3.3-39	ITS Table 3.3.7-1 is changed to be consistent with CTS Table 3.3-3. The Actuation Logic was split to reflect the SSPS, with only MODE 1-4 Applicability, and BOP-ESFAS portions and associated SR requirements in the CTS.	No, not in CTS.	No, not in CTS.	Yes	Yes
3.3-40	Add "and setpoint adjustment" to ITS 3.3.1 Condition E, similar to the Note for Condition D.	Yes	Yes	Yes	Yes
3.3-41	ITS 3.3.1 Condition L is deleted to match the plant-specific design and the CTS for the Source Range Neutron Flux Function in MODES 3, 4, and 5 with the Rod Control System incapable of rod withdrawal and all rods fully inserted. Under these conditions, the source range instrumentation does not provide a Reactor Trip System Function. The source range channels provide only indication [and inadvertent boron dilution mitigation] when in this Applicability. Requirements related to the source range neutron flux channels in MODES 3, 4, and 5 when all rods are fully inserted and are not capable of being withdrawn have therefore been [moved to ITS 3.3.9. Footnote (f) of ITS Table 3.3-1 is added to Function 5 and revised accordingly].	No, see CN 3.3-123.	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-57	Not used.	N/A	N/A	N/A	N/A
3.3-58	Adds new ITS 3.3.2 Condition [N] to reflect CTS Table 3.3-3 ACTION Statement [24] on manual AFW [and manual MSIV closure] initiation.	Yes	No, not in CTS.	No, adopted ISTS format.	Yes
3.3-59	Adds new ITS 3.3.2 Condition [R] to reflect CTS Table 3.3-3 ACTION Statement [21] on BOP-ESFAS portion of AFW initiation.	No, not in CTS.	No, not in CTS.	Yes	Yes
3.3-60	Consistent with the design and CTS, Surveillance Requirement[s 3.3.2.3 and 3.3.2.7 are deleted as they are] not used by any Function listed in Table 3.3.2-1.	Yes	Yes	No, used for BOP-ESFAS.	No, used for BOP-ESFAS.
3.3-61	Change ITS SR 3.3.2.11 Frequency to ~~18~~ [24] months per CTS Table 4.3-2 Functional Unit [8.c], which is the ESFAS P-4 permissive. [Deleted the Note stating that verification of set point is not required per the CTS.]	Yes	Yes	Yes	Yes
3.3-62	Consistent with the CPSES design and CTS, isolation of the MSIVs also requires isolation of the associated upstream drip pot isolation valves.	No	Yes	No	No
3.3-63	Revise ITS Table 3.3.2-1 [Notes (b) and (g)] per CTS Table [3.3-3] Notes [# and ##]. This revision is a clarification to the operator that describes the circumstances under which the [Steamline Pressure Negative Rate - High, Steam Pressure-low, or Pressurizer Pressure-low functions may be or are] blocked [relative to the] P-11 permissive.	Yes	Yes	Yes	Yes
3.3-64	Revise ITS Table 3.3.2-1 Note (j) to exclude the MFRVs, consistent with CTS [3.7.1.6]. [ ]	No, already in CTS.	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-65	A Note is added to the steam generator water level - high-high trip function to reflect the CPSES design and CTS. In the CPSES design, only three channels of the four steam generator water level signals provide input to this trip function. Therefore, in order to satisfy the single failure criterion, if one of these three channels is used as input to the Steam Generator Water Level Control System, its associated bistable must be placed in the tripped state.	No	Yes	No	No
3.3-66	The DCPP-specific MODE 4 requirement of the CTS is retained and added to Table 3.3.2-1 for SI actuated by Containment Pressure High.	Yes	No	No	No
3.3-67	In PAMS, add CPSES-specific operability requirements and Required Actions for the T-hot and T-cold indications consistent with both the current licensing basis and the intent of NUREG-1431. If a T-hot indication is unavailable, equivalent information is available from the Core Exit Temperature indication which is also a RG 1.97 variable. Similarly, if a T-cold indication is unavailable, equivalent information may be derived through the use of the steam generator pressure and steam tables, because the RCS cold leg temperature closely follows the steam generator saturation temperature.	No	Yes	No	No
3.3-68	A DCPP-specific Note is added to state that CONDITION D is only applicable in MODES 1 and 2. A new CONDITION H is added to require entering MODE 3 if CONDITION B is not met when entered due to not meeting CONDITION D.	Yes	No	No	No
3.3-69	{The phrase "...that is not normally energized" is deleted per the CTS. All of the instrumentation listed is normally energized at power.}	Yes	No	No	No}
3.3-70	The PAM instrumentation list is modified to reflect the CPSES design and CTS.	No, see CN 3.3-71.	Yes	No, see CN 3.3-21.	No, see CN 3.3-21.

[illegible]

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-86	Surveillance Requirement 3.3.5.2 is revised to reflect the current CPSES plant design and licensing basis. A Note is added to SR 3.3.5.2 indicating that setpoint verification is not applicable for the performance of the TADOT. This verification is performed during Channel Calobrations (see SR 3.3.5.3).	No	Yes, see also CNs 3.3-31, 3.3-130, and 3.3-131.	No	No
3.3-87	Not used.	NA	NA	NA	NA
3.3-88	Revise ITS 3.3.9 to apply in MODE 2 only below P-6 and to reflect ACTION Statement 5.b per CTS Table 3.3-1.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
3.3-89	Revise COT in ITS SR 3.3.9.3 to add the 4 hour allowance from ITS SR 3.3.1.7.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
3.3-90	Exclude neutron detectors from CHANNEL CALIBRATION ITS SR 3.3.9.4 per CTS Table 4.3-1, Functional Unit 6, Note 4.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
3.3-91	Add CHANNEL CHECK and response time surveillances (ITS SR 3.3.9.1 and SR 3.3.9.5) per CTS Table 4.3-1, Functional Unit 6, Note 12.	No, not in CTS.	No, not in CTS.	No, not in CTS.	Yes
3.3-92	Adds SR 3.3.4.2 Note that the ASP controls for the TDAFW pump and SG ASDs are not required to be verified prior to entry into MODE 3, consistent with CTS SR 4.3.3.5.3.	No, adopted ISTS format.	No, not in CTS.	No, adopted ISTS format.	Yes
3.3-93	ITS 3.3.1 Condition V is deleted. It is not entered from Table 3.3.1-1 nor do the Bases clarify when it would be needed, raising the concern of misinterpretation. Condition V does not replace LCO 3.0.3 requirements to assess when the plant is outside the licensing basis.	Yes	Yes	Yes	Yes
3.3-94	ITS 3.3.4 is revised per CTS [3.3.3.5] with regard to [~~ASP~~ remote shutdown panel] controls.	Yes	Yes	No, adopted ISTS format.	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.3-131	ITS 3.3.5 Condition B is replaced with new Conditions B, C, D, and E. Condition C in the ISTS is changed to Condition F. The CPSES CTS have specific actions for the various bus undervoltage and degraded voltage function. These actions allow an appropriate amount of time to restore an inoperable channel or declare the associated power source or bus inoperable and take action to isolate an inoperable power source. These actions are a proper way to respond to the inoperable channels because the actions result in taking the Required Actions in ITS 3.8 associated with the affected power source or bus. The new Conditions match the Actions of the CTS.	No	Yes	No	No
3.3-132	The trip setpoints for the loss of power diesel generator start instrumentation are relocated to a licensee controlled document. This approach is consistent with a format allowed by a reviewer's note for the RTS and ESFAS instrumentation.	No -adopted ITS format.	Yes	No, adopted ITS format.	No, adopted ITS format.
3.3-133	This change revises ITS LCO 3.3.5 and SR 3.3.5.3 to include the DG start sequence delay timers from DCPP CTS Table 3.3-4.	Yes	No	No	No
3.3-134	This change is Wolf Creek specific to revise the NOTE in Condition K of ITS 3.3.2 consistent with CTS Table 3.3-3 Action 16 for Function 7b and Amendment 43 to provide 4 hours foran additional channel to be placed in bypass for surveillance testing of other channels.	No	No	Yes	No
3.3-135	A MODE change restriction has been added per the matrix discussed in CN 1-02-LS-1 of the ITS 3.0 package.	Yes	Yes	Yes	Yes
3.3-136	The TADOT perfromed under ITS SR 3.3.2.7 includes verification of relay setpoints since the trip actuating devices being tested are the same circuits tested under ITS SR 3.3.5.2.	No, adopted ISTS format.	No, adopted ISTS format.	Yes	Yes
{3.3-137	The Condition for Function 4.c is changed from Condition D to E consistent with the DCPP CTS.	Yes	No	No	No}

B-1
C

{Insert A}

(B) & (C)

Insert A (page 21 of 21, Section 3.3 / Enclosure 6B)

3.3-139	This change adds new SR 3.3.2.13 which is the performance of an 18 month TADOT. SR 3.3.2.8 is the performance of a TADOT every 24 months for DCPP.	Yes	No	No	No

DCL-97-106, LAR 97-09
ITS 3.4 ERRATA

(C)

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1600 cubic feet (90% of span) and two groups of pressurizer heaters each having a capacity of at least 150 kW and capable of being powered from an emergency power supply. 03-05-M

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer water level not within limit ~~otherwise inoperable~~, be in at least HOT STANDBY with the ~~Reactor trip breakers open~~ all rods fully inserted and the Rod Control System incapable of rod withdrawal within 6 hours and in HOT SHUTDOWN within the following 6 hours. 03-01-LS4

03-04-LS29

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified ~~by measuring heater group power~~ at least once per ~~92 days.~~ 18 months. 03-02-LG 03-03-LS28

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once ~~per 18 months~~ by transferring power from the normal to the emergency power supply and energizing the heaters.

each REFUELING INTERVAL

ACTION: (continued)

\# 1) Pressure isolation valves in the residual heat removal (RHR) flow path when in, or during transition to or from, the RHR mode of operation are excluded in MODE 4. 06-08-LS9

\#\# Each valve used to satisfy this action must have been verified to meet surveillance requirement 4.4.6.2.2. 06-09-LS10

SURVEILLANCE REQUIREMENTS 06-12-M

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

~~a. Monitoring the containment atmosphere particulate or gaseous radioactivity monitor at least once per 12 hours;~~ 06-13-LS12

~~b. Monitoring the containment structure sump inventory and discharge at least once per 12 hours;~~ 06-13-LS12

~~c.~~ a. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals at least once per 31 days when the Reactor Coolant System pressure is 2235 ± 20 psig with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4. 06-14-A

~~d.~~ b. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours~~.~~. ~~except when T_{avg} is being changed by greater than 6°F/hour or when diverting reactor coolant to the liquid holdup tank, in which cases the required inventory balance shall be performed within 12 hours after completion of the excepted operation; and~~ not required to be performed until 12 hours after establishment of steady state operation. 06-17-LG

~~e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.~~ 06-26-LS30 06-16-LS14

4.4.6.2.2 ~~As specified in Table 3.4-1,~~ Reactor Coolant System pressure isolation valves ~~shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve~~ shall be demonstrated OPERABLE by verifying leakage to be within its limit: 06-07-LG 06-20-A

a. At least once each REFUELING INTERVAL ~~Every refueling outage~~ during startup.

~~b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and~~ 06-19-TR3

~~c.~~ b. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve. After each disturbance of the valve, in lieu of measuring leak rate, leak-tight integrity may be verified by absence of pressure buildup in the test line downstream of the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

(C)

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.14.1	-------------------NOTES------------------ 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. -- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ ~~[~~ 2215 ~~]~~ psig and ~~≤ [2255]~~ psig.	In accordance with the Inservice Testing Program, B and ~~[18]~~ months 24 B ~~AND~~ ~~Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months except for valves 8701 and 8702.~~ 3.4-24 AND (continued)

BASES

ACTIONS (continued)

experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is ~~within~~ consistent with the safety analyses assumptions of ensuring that a steam bubble exists in the pressurizer. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to be at their design rating. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The Frequency of ~~92 days~~ 18 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

SR 3.4.9.3

~~This SR is not applicable if the heaters are permanently powered by Class 1E power supplies.~~

24

This SR demonstrates that the heaters can be manually transferred from the normal to the emergency power supply and energized. The Frequency of ~~18~~ months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.

REFERENCES

1. FSAR, Section 15.
2. NUREG-0737, November 1980.

BASES (continued)

APPLICABILITY (continued)	Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time ~~allows~~ is available for operator action to mitigate the event.
	The LCO ~~Applicability~~ is modified by two Notes. Note 1 states ~~stating~~ that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions. Note 2 states that more than one charging pump may be capable of injection into the RCS during, and up to 1 hour after swapping charging pump operation.

move this ¶ to LCO section

ACTIONS

A.1 and B.1

With one ~~two~~ or more SI ~~HPI~~ pumps or two CCPs capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

~~Required Action B.1 is modified by a Note that permits two charging pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.~~

The CCP and the PDP are capable of injecting into the RCS both operating alone or simultaneously. Their operation is limited to the conditions specified in the PTLR. The current limitations are based on RCS temperature as follows:

RCS Temperature	Allowable Charging Pumps Capable of Injecting into the RCS
Greater than 270°F	Two CCPs AND one PDP
Less than or equal to 270°F but greater than approximately 162°F	One CCP AND one PDP
Less than or equal to approximately 162°F but greater than approximately 134°F	One CCP OR one PDP
Less than or equal to approximately 134°F	One CCP OR one PDP AND ECCS charging injection flow path isolated

(Continued)

BASES

ACTIONS (continued)

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS ~~relief valve~~ Class I PORV to protect against overpressure events.

G.1

The RCS must be depressurized and a vent must be established within 8 hours when:

a. Both required RCS ~~relief valves~~ Class I PORVs are inoperable; or

b. A Required Action and associated Completion Time of Condition A, B, D, E, or F is not met; or

c. The LTOP System is inoperable for any reason other than Condition A, B, C, D, E, or F.

The vent must be sized ≥ 2.07 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of zero SI ~~[one] [HPI]~~ pumps ~~[and a maximum of~~ one ~~charging pumps]~~ CCP are verified ~~in~~capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and ~~locked out~~, their breakers open. Verification that each accumulator is isolated is only required when accumulator isolation is required as stated in Note 1 to the ~~Applicability~~ LCO.

The SI ~~[HPI]~~ pumps and one CCP ~~charging pump[s]~~ are rendered incapable of injecting into the RCS for example, through removing the power from the pumps by racking the breakers out under administrative control or by isolating the discharge of the pump by closed isolation valves with power removed from the operators or by a manual isolation valve secured in the closed position.

(Continued)

C

BASES

ACTIONS (continued)

B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, 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 MODE 3 within 6 hours and MODE 5 within 36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside containment. 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 system.

~~C.1~~

~~The inoperability of the RHR autoclosure interlock renders the RHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the RHR autoclosure interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the autoclosure.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. This method results in testing each valve separately. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every ~~18~~ 24 months, a typical refueling cycle. ~~if the plant does not go into MODE 5 for at least 7 days.~~ The ~~18~~ 24 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) O&M Code, ~~Section XI~~ Part 10 (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(Continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

Test pressures less than 2235 psig but greater than 150 psig are allowed for valves where higher pressures would tend to diminish leakage channel opening. Observed leakage shall be adjusted for actual pressure to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one half power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. ~~The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months.~~ In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

Testing is not required for the RHR suction isolation valves more frequently than ~~18~~ 24 months as these valves are motor operated with control room position indication, inadvertent opening interlocks and system high pressure alarms.

SR 3.4.14.2 and 3.4.14.3 NOT USED

[~~Verifying that the RHR system autoclosure interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR heat exchangers beyond 125% of their design pressure of [600] psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be <[425] psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The [18] month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.~~]

(Continued)

C

CHANGE NUMBER	JUSTIFICATION
3.4-23	ITS SR 3.4.12.3 is revised to be consistent with the requirements denoted in LCO 3.4.12 and Industry Traveler WOG-51, Rev. 1. This change clarifies that the SR (to verify the accumulators isolated) is only applicable when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.
3.4-24	Improved SR 3.4.14.1 requires testing of PIVs prior to entering MODE 2 whenever the unit has been in MODE 5 for seven days or longer. This surveillance frequency is not in the DCPP CTS. DCPP tests the valves on an ~~18~~ month Frequency which is acceptable based on valve type, interlocks, position indication and system alarm 24 functions. These valves meet the criteria for testing at least every ~~18~~ months but do not require additional testing based on when the unit is in cold shutdown. None of these valves change position during normal plant operation. There is little reason to anticipate leakage to initiate in mid cycle when they have not changed position. In conformance with the CTS, DCPP has chosen not to include this surveillance frequency of retesting the valves following an extended period of operation in MODE 5.
3.4-25	Consistent with the CTS, the gross activity limit is added to LCO 3.4.16 Condition B rather than its first reference being in SR 3.4.16.1. This change is also consistent with the treatment of DOSE EQUIVALENT I-131.
3.4-26	DCPP design designates two pressurizer PORVs to provide the function for low temperature overpressure protection. The valves have been designated as a Class I system. The PORVs are air operated. The air supply is not a Class I system, consequently, the design includes a back-up nitrogen Class I system to power the valves in the event the air supply is not available. The surveillance for the nitrogen back-up system is in the DCPP CTS but not present in NUREG-1431. Improved TS SR 3.4.11.3 has been added to provide assurance of the operation of the nitrogen gas supply to the PORVs. Methodology of verifying operability has been moved to the Bases.
3.4-27	TS 3.4.12 Applicability for MODE 6 has an additional qualification if the head closure bolts are not fully de-tensioned. With the bolts fully de-tensioned this TS is not Applicable because the lack of tension provides a pressure relief path across the vessel flange. This Applicability is in the current licensing basis.
3.4-28	This change has added the description of a secured open valve. This description is in use in the DCPP CTS and in plant procedures. This description is also consistent with that used in NUREG-1431 SR 3.5.2.2 and SR 3.5.2.5.
3.4-29	The definition for CHANNEL FUNCTIONAL TEST (CFT) would be retained from the current DCPP TS to the improved TS. CFT is in active use in numerous procedures in the plant. The CFT is used in applications for which the CHANNEL FUNCTIONAL TEST (COT) is not fully suitable. Although CFT and COT definitions appear similar, there is one important difference. Strict adherence to COT requirements includes quantitative adjustments as appropriate to bring setpoints into the desired range. This requirement for quantitative adjustment can not be satisfied in a reasonable manner on some components/sensors/channels due to their design. CFT however, is a qualitative test to determine functionality. A loss of function indicated by the CFT results in a notification to, following existing procedures, restore the functional performance. The CFT is in the current licensing basis.

DCL-97-106, LAR 97-09
ITS 3.5 ERRATA

(E)

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

2) Verifying that each accumulator isolation valve is open ~~and power is removed~~.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to ~~1% of tank volume~~ 5.6% of narrow range indicated level by verifying the boron concentration of the accumulator solution. This surveillance is not required when the volume increase makeup source is the RWST ~~and the RWST has not been diluted since verifying that the RWST boron concentration is equal to or greater than the accumulator boron concentration limit, and:~~ 01-08-A 01-07-LG

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is disconnected by sealing the breaker in the open position.

(C)

SURVEILLANCE REQUIREMENTS (continued)

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water ~~by venting the ECCS pump casings and accessible discharge piping high points~~, and — 02-16-LG

2) Verifying that each ECCS valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

~~c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:~~ — 02-09-LG

~~1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and~~

~~2) At least once daily of the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.~~

d. At least once ~~per 18 months~~ each REFUELING INTERVAL by ~~a~~ a visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion:

e. At least once ~~per 18 months~~ each REFUELING INTERVAL by:

1) Verifying that each automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to its correct position on an actual or simulated ~~Safety Injection~~ actuation ~~test~~ signal. — 02-17-A — 02-11-TR1

2) Verifying that each ~~of the following~~ ECCS pumps starts automatically upon receipt of ~~a Safety Injection~~ an actual or simulated actuation ~~test~~ signal. — 02-11-TR1

a) Centrifugal charging pump.

b) Safety Injection pump, and

c) Residual Heat Removal pump.

~~f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:~~ — 02-12-LG

(C)

~~1) Centrifugal charging pump ≥ 2400 psid.~~

~~2) Safety Injection pump ≥ 1455 psid, and~~

~~3) Residual Heat Removal pump ≥ 165 psid~~.

Verifying, in accordance with the Inservice Testing Program, that each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head. 02-12-LG

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

~~1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and~~ 02-13-TR3

2) At least once ~~per 18 months~~ each REFUELING INTERVAL.

Charging Injection Throttle Valves	Safety Injection Throttle Valves
8810A	8822A
8810B	8822B
8810C	8822C
8810D	8822D

~~h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that~~: 02-15-LG

~~1) For centrifugal charging pumps, with a single pump running:~~

~~a) The sum of injection line flow rates, excluding the highest flow rate, is greater than or equal to 299 gpm, and~~

~~b) The total flow rate through all four injection lines is less than or equal to 461 gpm, and~~

~~c) The difference between the maximum and minimum injection line flow rates is less than or equal to 15.5 gpm, and~~

C

CHANGE NUMBER	NSHC	DESCRIPTION
03-04	LG	Consistent with NUREG-1431, the ACTION b terminology is revised. The requirement to restore at least one ECCS subsystem is revised to "immediately initiate action to restore" a residual heat removal (RHR) subsystem. With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, when the only available heat removal system is the RHR. Therefore, the appropriate ACTION is to initiate measures to restore one ECCS RHR subsystem and to continue the ACTIONS until the subsystem is restored to OPERABLE status. Also, the alternate requirement (if RHR cannot be restored) to maintain T_{avg} <350°F by use of alternate heat removal methods is descriptive information and is moved to the Bases. The transition to MODE 3 is already prohibited in this scenario by the ECCS specification for MODES 1, 2, and 3.
03-05	TR 2	Consistent with NUREG-1431, the requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted. This change is acceptable because the requirement to submit a report is sufficiently addressed by the reporting requirements contained in 10 CFR 50.73.
03-06	A	Consistent with TSTF-90, a Note is added to the LCO that clarifies an RHR train's ECCS function is operable if it is capable of being manually realigned to the ECCS mode of operation. This is an administrative change to provide clarification.
03-07	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-08	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-09	M	This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
03-10	LS 6	Consistent with NUREG-1431, the requirement to demonstrate ECCS train OPERABILITY in MODE 4 in SR [4.5.3.1] has been revised to delete the 31 day surveillance to verify the correct position of each valve in the ECCS flow path which is not already locked in place, and the ~~18~~ [24] month surveillance to verify automatic actuation of ECCS pumps and automatic valves.
03-11	LG	The minimum RCS temperature limit below which the CCP and SI pumps must be demonstrated not capable of injecting into the RCS is replaced by the statement "below the temperature where LTOP is required as specified in the pressure temperature limits report (PTLR)." The minimum temperature is a plant specific requirement based on the reactor vessel material characteristics documented in the PTLR and is periodically reviewed and adjusted as required. Referring to the PTLR for the current value is consistent with the relocation of the pressure temperature limits from ITS Section 3.4.3 to

[illegible]

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
03-05 TR 2	The requirement to submit a special report within 90 days of an ECCS actuation and injection event is deleted.	Yes	Yes	Yes	Yes
03-06 A	A note is added to the LCO that clarifies an RHR train's ECCS function is OPERABLE if it is capable of being manually realigned to the ECCS mode of operation.	Yes	Yes	Yes	Yes
03-07 M	The surveillance frequency to verify a maximum of one CCP capable of injecting into the RCS is changed from "at least once per 31 days thereafter," to "at least once per 12 hours thereafter."	No, the once per 12 hour surveillance frequency is in the CTS.	No, see CN 03-09-M.	Yes	Yes
03-08 A	A footnote is added to SR [4.5.3.1.1] indicating that the CTS SR to verify the RHR interlock action is not applicable when the RHR suction isolation valves are open to satisfy LCO [3.4.8.3].	No, not in CTS.	Yes	Yes	Yes
03-09 M	The surveillance frequency to verify a maximum of two CCPs capable of injecting into the RCS is changed from within 4 hours after entering MODE 4 from MODE 3 or prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, whichever occurs first and once per 31 days, to once per 12 hours.	No, not in CTS.	Yes	No, see CN 03-07-M.	No. see CN 03-07-M.
03-10 LS 6	This change deletes the 31 day surveillance to verify position of valves in the ECCS flow path, and the ~~18~~ [24] month surveillance to verify automatic actuation of ECCS pumps and automatic valves. (MODE 4)	Yes	Yes	Yes	Yes

Ⓔ

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

PAGE

E

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-4
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

In conformance with NUREG-1431 Rev. 1, the requirement for having the charging pumps/safety injection pumps 'inoperable' has been revised to preclude injection into the RCS. This change is consistent with the cold overpressure analysis requirements. This change results in the operability statements being revised and allows deletion of the notes which were in place for testing or accumulator filling. This change is less restrictive on the configuration of the centrifugal charging and safety injection pumps but does not result in a less conservative operational position as flow to the RCS is still precluded.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> 1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> 2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> 3. *Involve a significant reduction in a margin of safety."*

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

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

The intent of specifying that the required number of centrifugal charging pumps/safety injection pumps be inoperable is to preclude the possibility of injecting flow into the RCS in excess of that analyzed for the low temperature overpressure protection system. The method of precluding flow is inconsequential to the probability or consequences of any previously evaluated accident.

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

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

(E)

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS-4
(continued)

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

The proposed change does not affect the acceptance criteria for any analyzed event. The margin of safety established by the LCOs also remains unchanged. Thus there is no reduction in the margin of safety from that previously established.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS-4" resulting from the conversion to the improved TS format satisfy the no significant hazards consideration standards of 10 CFR 50.92(c); and accordingly, a no significant hazards consideration finding is justified.

C

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS6

10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

[24]

Consistent with NUREG-1431, the requirement to demonstrate ECCS train OPERABILITY in MODE 4 has been revised to delete the 31 day surveillance to verify the correct position of each valve in the ECCS flow path which is not already locked in place, and the ~~18~~ month surveillance to verify automatic actuation of ECCS pumps and automatic valves. This change is acceptable because the ECCS operational requirements are reduced due to the stable reactivity conditions and limited core cooling requirements associated with operation in MODE 4 and the unlikelihood of occurrence of a DBA in MODE 4. It is understood these surveillance reductions increase the possibility that automatic positioning of ECCS valves and actuation of ECCS pumps may not occur on a SI actuation signal. In MODE 4, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

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

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> 1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> 2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> 3. *Involve a significant reduction in a margin of safety."*

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

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

   Due to the stable conditions associated with operation in MODE 4, the probability of occurrence of a DBA is lower than in MODES 1, 2, and 3. Because of the reduced core cooling requirements, sufficient time is available for manual actuation/alignment of required ECCS equipment. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

C

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.5.2.2	Verify each ECCS 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	
[SR 3.5.2.3	Verify ECCS piping is full of water.	31 days]	B
SR 3.5.2.4	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program	
SR 3.5.2.5	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[18~~ months] 24	B
SR 3.5.2.6	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	~~[18~~ months] 24	B
SR 3.5.2.7	Verify, for each ECCS throttle valve listed below, each mechanical position stop is in the correct position. Charging Injection Throttle Valves: 8810A, 8810B, 8810C, 8810D Safety Injection Throttle Valves: 8822A, 8822B, 8822C, 8822D	~~18~~ months 24	3.5-3 B-PS
SR 3.5.2.8	Verify, by visual inspection, each ECCS train containment recirculation sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	~~[18]~~ months 24	B PS

(E)

BASES

BACKGROUND (continued)

This interlock also prevents inadvertent closure of the valves during normal operation prior to an accident. However, if these valves were closed, they would be automatically opened as a result of an SI signal. These features ensure that the valves meet the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 (Ref. 1) for "operating bypasses" and that the accumulators will be available for injection without reliance on operator action.

The accumulator size, water volume, and nitrogen cover pressure are selected so that three of the four accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. The need to ensure that three accumulators are adequate for this function is consistent with the LOCA assumption that the entire contents of one accumulator will be lost via the RCS pipe break during the blowdown phase of the LOCA.

APPLICABLE SAFETY ANALYSES

The accumulators are assumed OPERABLE in both the large and small break LOCA analyses at full power (Ref. 2 and 4). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the accumulators. Reference to the analyses for these DBAs is used to assess changes in the accumulators as they relate to the acceptance limits.

The limiting large break LOCA is a double ended guillotine break in the RCS piping. ~~at the discharge of the reactor coolant pump.~~ During this event, the accumulators discharge to the RCS as soon as RCS pressure decreases to below accumulator pressure.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of ECCS flow with no credit taken for ECCS pump flow until an effective delay has elapsed. In the early stages of a LOCA, with or without a loss of offsite power, the accumulators provide the sole source of makeup water to the RCS. The assumption of loss of offsite power is required by regulations and conservatively imposes a delay wherein the ECCS pumps cannot deliver flow until the emergency diesel generators start, come to rated speed, and go through their timed loading sequence. The delay time is conservatively set with an additional 2 seconds to account for SI signal generation. In cold leg break scenarios, the entire contents of one accumulator are assumed to be lost through the break. No operator action is assumed during the blowdown stage of a large break LOCA.

(E)

BASES

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2,.and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem.

Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal. ~~and initiating semi-automatic switchover of suction having its~~ During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold legs. The ECCS suction is manually transferred to the containment recirculation sump to place the system in the recirculation mode of operation to supply its flow to the RCS hot and cold legs. During the recirculation operation, the RHR pumps provide suction to the charging and SI pumps.

~~During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment recirculation sump and to supply its flow to the RCS hot and cold legs.~~

During recirculation operation, the flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

As indicated in Note 1, the SI flow paths amy be isolated for 2 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. The flow path is readily restorable from the control room and a single active failure (Ref. 7) is not assumed coincident with this testing. Therefore the ECCS trains are considered OPERABLE during this isolation.

move to Applicability Section

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

(continued)

C

BASES

REQUIREMENTS
(continued)

~~Section XI of~~ the ASME Code. (Ref. 8) This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is ~~greater than or equal to~~ within the performance assumed in the plant safety analysis. SRs are specified in ~~Technical Requirements Manual and in~~ the applicable portions of the Inservice Testing Program, which encompasses ~~Section XI~~ Part 6 of the ASME Code for Operation and Maintenance of Nuclear Power Plants. (Ref. 8). ~~Section XI~~ This section of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

SR 3.5.2.5 and SR 3.5.2.6

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The ~~18~~ 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The ~~18~~ 24 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.7

The correct ~~Realignment~~ position of throttle/runout valves in the ECCS flow paths ~~on an SI signal~~ is necessary for proper ECCS performance. These manual throttle/runout valves are positioned during flow balancing and have mechanical locks and seals ~~-stops~~ to ~~allow~~ ensure that the proper positioning for restricted flow to a ruptured cold leg~~, ensuring~~ is maintained. The verification of proper position of a throttle/runout valve can be accomplished by confirming the seals ~~and lock~~ have not been altered since the last performance of the flow balance test. Restricting the flow to a ruptured cold leg ensures ~~and~~ that the other cold legs receive at least the required minimum flow. ~~This Surveillance is not required for~~

(C)

SURVEILLANCE REQUIREMENTS (continued)

~~plants with flow limiting orifices.~~ The ~~18~~ 24 month Frequency is based on the same reasons as those stated in SR 3.5.2.5 and SR 3.5.2.6.

SR 3.5.2.8

Periodic inspections of the containment recirculation sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The ~~18~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

Opening the containment recirculation sump access hatch in MODES 1 through 4 is considered to be a condition which is outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered. (Ref. 9)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 35.
2. 10 CFR 50.46.
3. FSAR, Sections 6.3 and 7.3.
4. FSAR, Chapter 15, "Accident Analysis."
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.
7. BTP EICSB-18, Application of the Single Failure Criteria to Manually-Controlled Electrically-Operated Valves.
8. ASME/ANSI OM-1987, "Operational Maintenance of Nuclear Power Plants", including OM-a-1988 addenda, Part 6, "Inservice Testing of Pumps in Light Water Reactor Power Plants," and part 10, "Inservice Testing of Valves in Light Water Reactor Power Plants."
9. NRC letter to PG&E, EA 89-241, April 5, 1990, CHRON 148598.

(E.)

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS – Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2. and subsequently transferring RHR pump suction to the containment recirculation sump.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable core reactivity and the lower heat removal conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuations ~~is~~ are not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA. (Ref. 1)

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation.

The ECCS trains satisfy Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii).

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains (as defined for MODE 4) is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

The LCO is modified by a Note that allows a RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned.

2

duplicative of last ¶

E

BASES

LCO (continued)

~~(remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.~~

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment recirculation sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS charging and RHR pumps and their respective supply headers to each of the four cold legs. ~~injection nozzles~~. In the long term, this flow path may be switched to take its supply from the containment recirculation sump and to deliver its flow to the RCS hot and cold legs.

This LCO is modified by a Note that allows an RHR train to be considered OPERABLE during system alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS high head and low head train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

ACTIONS (continued)

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat

Ⓔ

ENCLOSURE 6B

CONVERSION COMPARISON TABLE - NUREG-1431

Conversion Comparison Table (1 Page)

Replace
Entire Section
with this!

(Enclosure 6B for 3.7 was incorrectly inserted into the 3.5 package).

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.5-1	Replaced "pressurizer pressure" with "RCS pressure."	Yes	Yes	Yes	Yes
3.5-2	The Completion Time of LCO 3.5.1, Condition B, is changed from 1 hour to 24 hours to reflect the CTS.	No, not part of the CTS.	No, not part of the CTS.	Yes, license Amendment pending.	Yes, CTS per OL Amendment No. 91.
3.5-3	Adds the word" mechanical" with regard to throttle valve position stop consistent with the CTS.	Yes	Yes	Yes	Yes
3.5-4	This change increases the RCP seal flow Completion Time from 4 to 72 hours, with a new added verification that at least 100 percent of the assumed charging flow remains available.	Yes	Yes	No, LCO 3.5.5 is not applicable.	No, LCO 3.5.5 is not applicable.
3.5-5	Deleted reference to CCP discharge header pressure to reflect CTS.	Yes	Yes	No, not part of the CTS.	No, not part of the CTS.
3.5-6	SR 3.5.3.1 Note is moved to LCO per Traveler TSTF-90.	Yes, per LAR 96-03.	Yes	Yes	Yes
3.5-7	Not used.	N/A	N/A	N/A	N/A
3.5-8	Moves the Notes from the "APPLICABILITY" to the "LCO.". Also revises the wording in Note 2 from "declared inoperable" to "made incapable of injecting."	No, not part of CTS.	Yes	Yes	Yes
3.5-9	The seal injection/return valves (BGV0198-BGV0202) are included in ITS SR 3.5.2.7 since they are included in CTS 4.5.2.g.2.	No, not part of the CTS.	No, not part of the CTS.	Yes	Yes

DCL-97-106, LAR 97-09
ITS 3.6 ERRATA

C

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM 07-01-A

LIMITING CONDITION FOR OPERATION

3.6.1.7 One purge supply line and/or one purge exhaust line of the Containment Purge System may be open or the vacuum/pressure relief line may be open ~~. The vacuum/pressure relief line may be open~~ provided the vacuum/pressure relief isolation valves are blocked to prevent opening beyond 50° (90° is fully open). Operation with any two of these three lines open is permitted. ~~Operation with the purge supply and/or exhaust isolation valves open or with the vacuum/pressure relief isolation valves open up to 50° shall be limited to less than or equal to 200 hours during a calendar year.~~ 07-04-R

APPLICABILITY: MODES 1, 2, 3, and 4. 11-02-A

ACTION: ** *** 03-13-A

~~With a containment purge supply and/or exhaust isolation valve open or the vacuum/pressure relief isolation valves open up to 50° for more than 200 hours during a calendar year or the Containment Purge System open and the vacuum/pressure relief lines open, or with the vacuum/pressure relief isolation valves open beyond 50°~~ With two containment purge supply or exhaust valves or two vacuum/pressure relief valves on the same penetration inoperable for reasons other than leakage, close the open isolation valve(s) or isolate the penetration(s) flowpath(s) within 1 hour; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. 07-04-R 07-05-A 11-12-A

(new) One or more penetration flow paths with one or more containment purge or vacuum/pressure relief valves not within purge valve leakage limits. Within 24 hours isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. Verify the affected penetration flow path is isolated once per 31 days for isolation devices outside containment and prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment* and perform Surveillance 4.6.3.4 for the resilient seal purge valves closed to comply with this Required Action ~~E.1~~D.1, once per 92 days. 07-02-LS9 01-04-LS1

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The position of the containment purge supply and exhaust isolation valves and the vacuum/pressure relief isolation valves shall be determined closed at least once per 31 days except for one valve in a penetration flow path while in action 3.6.1.7 for excessive leakage. 07-03-A

~~4.6.1.7.2 The cumulative time that the purge supply and/or exhaust isolation valves or the vacuum/pressure relief isolation valves have been open during a calendar year shall be determined at least once per 7 days.~~ 07-04-R

4.6.1.7.3 The 12-inch vacuum/pressure relief isolation valves shall be verified to be blocked to prevent opening beyond 50° at least once ~~per 18 months~~ each refueling interval. ED

* Isolation devices in high radiation areas may be verified by use of administrative means. 01-04-LS1

** Separate Condition entry is allowed for each penetration flow path. 11-02-A

*** Enter applicable Conditions and Required Actions of the "Containment" LCO when leakage results in exceeding the overall containment leakage rate. 03-13-A

C

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS 08-04-A

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION 8-01-LG

3.6.2.1 Two Containment Spray Systems shall be OPERABLE ~~with each Spray System capable of taking suction from the RWST and transferring spray function to a RHR System taking suction from the containment sump.~~

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: * 08-11-LS2

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; ~~restore the inoperable Spray System to OPERABLE status within the next 48 hours or~~ and be in COLD SHUTDOWN within the following ~~30~~ 78 hours. 08-02-A

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying that ~~on recirculation flow~~, each pump's developsed head at the flow test point is ~~a differential pressure of~~ greater than or equal to ~~205 psid~~ the required developed head when tested pursuant to ~~Specification 4.0.5~~ the Inservice Test Program; 08-08-LG

c. At least once ~~per 18 months~~ each REFUELING INTERVAL by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal, and

   2) Verifying that each spray pump starts automatically on an actual or simulated actuation signal.

d. At least once per 10 years by ~~performing an air or smoke flow test through each spray header and~~ verifying each spray nozzle is unobstructed. 08-06-LG

* Additionally, a completion time of 10 days from discovery of failure to meet the conditions of 3.6.2.1 and 3.6.2.3. 08-11-LS2

(C)

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE ~~with~~:

a. A spray additive tank with a contained volume of between ~~2025 and 4000 gallons~~ 46.2 % and 91.9% of between 30 and 32% by weight NaOH solution, and 09-01-A

b. ~~Two spray additive eductors each capable of adding NaOH solution from the chemical additive tank to a Containment Spray System pump flow.~~ 09-02-LG

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

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours. 09-03-A

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 6 months by:

1) Verifying the contained solution volume in the tank, and

2) Verifying the concentration of the NaOH solution by chemical analysis.

c. At least once each REFUELING INTERVAL ~~per 18 months~~ by verifying that each automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to its correct position on ~~a~~ an actual or simulated ~~Containment Spray actuation test~~ signal; and 09-04-A 09-05-TR1

d. At least once per 5 years by verifying both spray additive and RWST full flow from the test valve 8993 through each solution flow path in the Spray Additive System. 09-07-M

(C)

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued

2) Verifying a cooling water flow rate of greater than or equal to 1650* gpm to each cooler, and — 10-03-LG

3) Verifying that each containment fan cooler unit starts on low speed.

b. At least once ~~per 18 months~~ each REFUELING INTERVAL by verifying that each containment fan cooler unit starts automatically on ~~a Safety Injection test~~ an actual or simulated actuation signal. — 09-05-TR1

~~* The CFCU cooling water flow rate requirement of TS 4.6.2.3a.2) may not be met during Section XI testing and in Mode 4 during residual heat removal heat exchanger operation.~~ — 10-03-LG

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each automatic containment isolation valve that is not locked, sealed or otherwise secured in position shall be demonstrated OPERABLE at least once ~~per 18 months~~ EACH REFUELING INTERVAL by: 11-14-A 11-07-LG

a. Verifying that on ~~a Phase "A" Isolation test~~ an actual or simulated signal, each ~~Phase "A" isolation~~ valve actuates to its isolation position. 11-08-TR1

~~b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and~~ 11-07-LG

~~c. Verifying that on a Containment Ventilation Isolation test signal, each containment ventilation isolation valve actuates to its isolation position.~~ 11-07-LG

Note 2: ** Separate Condition entry is allowed for each penetration flow path. 11-02-A

*** Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves. 11-03-A

**** Enter applicable Conditions and Required Actions of Specification 3.6.1.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria. 03-13-A

(C)

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent Hydrogen Recombiner Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Recombiner System inoperable*, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours. 13-05-LS23

(new) With two hydrogen recombers inoperable, verify within 1 hour and once per 12 hours thereafter, by administrative means, that the hydrogen control function is maintained, and restore one hydrogen recombiner to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours. 13-01-LS17

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

a. At least once each REFUELING INTERVAL ~~refueling interval~~ ~~18 months~~ by ~~verifying, during~~ performing a Recombiner System functional test. ~~that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 kW; and~~ 13-02-LS18 13-03-LG

b. At least once each REFUELING INTERVAL ~~refueling interval~~ ~~18 month~~ by: 13-02-LS18

~~1) Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits~~. 13-04-LG

2) Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure ~~(i.e., loose wiring or structural connections, deposits of foreign materials, etc.)~~, and 13-03-LG

3) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test ~~following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms~~. 13-03-LG

* 3.0.4 is not applicable. 13-05-LS23

C

DESCRIPTION OF CHANGES TO TS SECTION 3/4.6
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
12-05	LS16	Revises the Frequency of the hydrogen monitor surveillance to perform CHANNEL CALIBRATION from 92 days on a staggered test basis to once per 18 months consistent with NUREG-1431. The hydrogen monitors are part of the PAM instrumentation and their primary function is to detect high hydrogen concentration conditions that may occur during accident situations. This change is acceptable because the primary means of reducing hydrogen concentration during accidents is via the independent hydrogen recombiners [and hydrogen purge systems]. Failure of the monitors would not affect the capabilities of [these systems]. Further changing the CHANNEL CALIBRATION surveillance interval from 92 days (on a staggered test basis) to every [18 months] is not expected to effect the reliability or performance of the hydrogen monitors based on industry operating experience.
12-06	LG	The details provided for performing the CHANNEL CALIBRATION are moved out of the SR. This information is procedural in nature and is not consistent with the level of detail in NUREG-1431. The information is moved to the Bases for ITS SR 3.3.3.2.
12-07	M	A new SR is added for DCPP requiring a CHANNEL CHECK every 31 days (if energized) for the hydrogen analyzer/ monitors. This change is consistent with NUREG-1431.
13-01	LS17	A new Condition has been added to this specification. This Condition describes the Required Action for two hydrogen recombiners inoperable. Whereas in the current specification LCO 3.0.3 applied, this change allows up to 7 days to restore one hydrogen recombiner to OPERABLE status, based on the availability of the containment hydrogen purge system to provide the required safety function. In order to use this ACTION time, the Required Actions require that the hydrogen control function be verified available within 1 hour and once every 12 hours thereafter. This administrative verification will assure that the hydrogen purge system is capable of performing the safety function if an event occurs. Also, the Bases for operation of the recombiners indicates that if a design basis event occurs, 8 days or more would elapse before the containment atmosphere approached the lower flammability limit for hydrogen. Therefore, it is reasonable to assume that the inoperability of two hydrogen recombiners will not significantly jeopardize the capability of the facility to respond to a design basis event. This change is consistent with NUREG-1431.
13-02	LS18 This change is not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).	~~The current SR to perform a hydrogen recombiner functional test every 6 months is revised to every 18 months consistent with NUREG-1431. This change is considered acceptable due to the redundancy and proven high reliability of the system. Hydrogen recombiner operating experience has shown that functional test failures are rare. In addition, the fully redundant and independent hydrogen purge system provides an alternate, and equally effective, method of controlling hydrogen. The proposed change is in accordance with NUREG-1366, "Improvement to Technical Specification Requirements" and NUREG-1431.~~
13-03	LG	Descriptive information regarding the current hydrogen recombiner surveillances is moved into the Bases. The proposed changes to the surveillances are consistent with the wording and detail present in the NUREG-1431 surveillance requirements.

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CHANGE NUMBER	NSHC	DESCRIPTION
13-04	LG	The SR to perform a CHANNEL CALIBRATION on all the hydrogen recombiner instrumentation is moved to a Licensee controlled document in accordance with NUREG-1431. These calibrations and any necessary compensatory measures (i.e., substitute test instrumentation) will be controlled administratively by the plant preventive maintenance and operational procedures. This change is acceptable based on the system redundancy, available alternate means of controlling hydrogen, the fact the recombiners are controlled manually, and the instrumentation does not provide essential control or interlock function. In addition, the functional test required by the TS every ~~18~~ [24] months verifies the operation of the hydrogen recombiner instrumentation. This change is consistent with NUREG-1431.
13-05	LS23	Added statement that LCO 3.0.4 is not applicable to ACTION. This allowance is based upon the pressure of another 100% hydrogen recombiner, the hydrogen purge system and the time available for operator action after a LOCA.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-02 M	The MODE of Applicability for the hydrogen monitors is extended to MODE 3.	Yes	Yes	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
12-03 LS15	The ACTION is revised to require a special report to be submitted within 14 days in lieu of being in HOT STANDBY within 6 hours, if one train of hydrogen monitoring cannot be restored to OPERABLE within 30 days.	Yes	Yes	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
12-04 M	Adds the requirement to be in HOT SHUTDOWN within 12 hours if both trains of hydrogen monitoring are inoperable and one train was not restored within 72 hours.	Yes	Yes	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
12-05 LS16	Revises the frequency of the surveillance to perform CHANNEL CALIBRATION from 92 days on a staggered basis to once per 18 months.	Yes	No, CTS requirement redefined (see 12-06-LG).	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
12-06 LG	The details provided for performing the CHANNEL CALIBRATION are moved out of the SR. The information is moved to the Bases.	Yes	Yes	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
12-07 M	A new SR is added for DCPP requiring a CHANNEL CHECK every 31 days for the hydrogen analyzer/ monitors.	Yes	No, SR already in CTS.	No, CTS hydrogen monitoring requirements are not in this Section.	No, CTS hydrogen monitoring requirements are not in this Section.
13-01 LS17	A new Condition has been added to this specification. This condition describes the Required Action for two hydrogen recombiners inoperable.	Yes	Yes	Yes	Yes
13-02 LS18	The current SR to perform a hydrogen recombiner functional test every 6 months is revised to every 18 months.	~~Yes,~~ No CTS ~~required~~ already has Refueling Interval which ~~changed to 18~~ is defined as 24 months.	Yes	Yes	No, CTS already has 18 month.

C

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

(C)

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS18
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Not Applicable to DCPP

The current surveillance requirement to perform a hydrogen recombiner functional test every 6 months is revised to every 18 months consistent with NUREG-1431. This change is considered acceptable due to the redundancy and proven high reliability of the system. Hydrogen recombiner operating experience has shown that functional test failures are rare. In addition, the fully redundant and independent hydrogen purge system provides an alternate, and equally effective, method of controlling hydrogen. The proposed change is in accordance with NUREG-1366, "Improvement to Technical Specification Requirements," and NUREG-1431.

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

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21 (b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> 1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> 2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> 3. *Involve a significant reduction in a margin of safety."*

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

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

   The proposed change, to extend the surveillance interval for the hydrogen combiner functional tests, does not result in any hardware changes. The hydrogen recombiners are not assumed in the initiation of any analyzed event. Their role is in reducing hydrogen concentration in containment, and thereby limiting potential accident consequences. Hydrogen recombiner operating experience has shown that functional test failures are rare. Thus, the extended surveillance interval will not result a loss in the capability to reduce hydrogen concentration in containment. Additionally, in the unlikely event both recombiners fail, a diverse method of reducing hydrogen concentration is available utilizing the containment purge system.

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

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

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

   The increased interval between hydrogen recombiner functional tests is acceptable based on the relative simplicity of the recombiner system and industry experience which indicates that the recombiner availability can be assured with reduced testing. As a result, any reduction in a margin of safety will be insignificant.

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IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC ~~LS18~~
(continued)

Not Applicable to DCPP

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS18" resulting from the conversion to the ITS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

(C)

SURVEILLANCE REQUIREMENTS (Continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.3.7	----------NOTE---------- This surveillance is not required when the penetration flow path is isolated by a leak tested blank flange. ---------------------------- Perform leakage rate testing for containment purge supply and exhaust and vacuum/pressure relief valves with resilient seals in accordance with the Containment Leakage Rate Testing Program.	3.6-13 184 days AND Within 92 days after opening the valve 3.6-17
SR 3.6.3.8	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	~~18~~ 24 months
SR 3.6.3.9	NOT USED ~~Cycle each weight or spring loaded check valve not testable during operation through one complete cycle of full travel, and verify each check valve remains closed when the differential pressure in the direction of flow is ≤ [1.2] psid and opens when the differential pressure in the direction of flow is ≥ [1.2] psid and ≤ [5.0] psid.~~	~~18 months~~ B-PS
SR 3.6.3.10	Verify each 12 inch containment ~~purge valve~~ vacuum/pressure relief valve is blocked to restrict the valve from opening > 50°.	~~18~~ 24 months B-PS

X

(continued)

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
SR 3.6.6A.2	Operate each required ~~containment cooling train fan unit~~ CFCU for ≥ 15 minutes.	31 days PS
SR 3.6.6A.3	Verify ~~each [required] containment cooling train~~ component cooling water flow rate to each required CFCU is ≥ 1650 gpm.	31 days PS
SR 3.6.6A.4	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6A.5	Verify each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~18~~ 24 months B
SR 3.6.6A.6	Verify each containment spray pump starts automatically on an actual or simulated actuation signal.	~~18~~ 24 months B
SR 3.6.6A.7	Verify each ~~[required] containment cooling train~~ CFCU starts automatically on an actual or simulated actuation signal.	~~18~~ 24 months 3.6-14 B

(C)

SURVEILLANCE REQUIREMENTS (Continued)

SURVEILLANCE		FREQUENCY
SR 3.6.7.2	Verify spray additive tank solution volume is ≥ ~~[2568] gal~~ 46.2% and ≤ ~~[4000] gal~~ 91.9%.	184 days 3.6-10 B-PS
SR 3.6.7.3	Verify spray additive tank NaOH solution concentration is ≥ 30% and ≤ 32% by weight.	184 days B-PS
SR 3.6.7.4	Verify each spray additive automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~18~~ 24 months B
SR 3.6.7.5	Verify spray additive flow ~~[rate]~~ from each solution's flow path.	5 years B-PS

C

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.8.1	Perform a system functional test for each hydrogen recombiner.	~~18~~ 24 months B
SR 3.6.8.2	Visually examine each hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	~~18~~ 24 months B
SR 3.6.8.3	Perform a resistance to ground test for each heater phase.	~~18~~ 24 months B

BASES (Continued)

SURVEILLANCE REQUIREMENTS

containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. ~~Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only not normally challenged when the containment air lock door is opened used for entry and exit (procedures require strict adherence to single door opening).~~ This test is only required to be performed ~~upon entering or exiting a containment air lock but is not required more frequently than every 184 days~~ every 24 months. The 24 month Frequency is based on avoiding the loss of containment OPERABILITY if the Surveillance ~~were performed with~~ the reactor at power~~. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. The 184 day 24 month Frequency is based~~ and on engineering judgement ~~and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel given that the interlock is not challenged during use of the airlock~~.

is (eg, both doors are open simultaneously) failed while

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Section 3.8, 6.2, and 15.

BASES

SR 3.6.3.8

Automatic containment isolation valves close on a Phase A, Phase B, or CVI signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic ~~containment isolation~~ valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The ~~18~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the ~~18~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.9 Not Used

~~In subatmospheric containments, the check valves that serve a containment isolation function are weight or spring loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.9 verifies the operation of the check valves that are not testable during unit operation. The Frequency of 18 months is based on such factors as the inaccessibility of these valves, the fact that the unit must be shut down to perform the tests, and the successful results of the tests on an 18 month basis during past unit operation.~~

SR 3.6.3.10

~~Reviewer's Note: This SR is only required for those units with resilient seal purge valves allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices on the valves that are not permanently installed.~~

Verifying that each ~~[42]~~ 12 inch containment ~~purge~~ pressure/vacuum relief valve is blocked to restrict opening to ~~≤ [50]%~~ 50° is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the ~~purge~~ containment pressure/vacuum relief valves must close to maintain containment leakage within the values assumed in the accident analysis. ~~At other times when purge valves are required to be capable of closing (e.g., during movement of recently irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open.~~ The ~~18~~ 24 month Frequency is appropriate because the blocking devices are not typically removed ~~only during a refueling outage~~ except during maintenance.

(Continued)

E

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4A Containment Pressure ~~(Atmospheric, Dual, and Ice Condenser)~~

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer modeled pressure transients. The worst case LOCA generates larger mass and energy release than the worst case SLB (SLB at 30% power) generates the greatest mass and energy release rate. Thus, the ~~LOCA~~ SLB event bounds the ~~SLB~~ LOCA event from the containment peak pressure standpoint (Ref. 1).

The initial pressure condition used in the containment analysis was ~~17.7~~ 16 psia (~~3.0~~ 1.3 psig). This resulted in a maximum peak pressure from a ~~LOCA of 53.9~~ SLB of 42.25 psig. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a, results from the limiting ~~LOCA~~ SLB at 30% power. The maximum containment pressure resulting from the worst case ~~LOCA, 44.1~~ SLB, 42.25 psig, does not exceed the containment design pressure, ~~55~~ 47 psig. X

The containment was also designed for an external pressure load equivalent to -3.5 ~~2.5~~ psig. The inadvertent actuation of the Containment Spray System was

(Continued)

BASES (Continued)

normal tests of centrifugal pump performance required by ~~Section XI~~ Part 6 of the ASME O&M Code (Ref. ~~8~~ 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.6A.5 and SR 3.6.6A.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment ~~"S" (High 1) 3~~ high-high pressure signal with a coincident "S" signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. ~~The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power.~~ Operating experience has shown that these components usually pass the Surveillances when performed at the ~~18~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

~~The surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance may be used to satisfy both requirements.~~

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6A.7

This SR requires verification that each ~~[required] containment cooling train~~ CFCU actuates upon receipt of an actual or simulated safety injection signal. The ~~[18]~~ 24 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the ~~18~~ 24 month Frequency.

SR 3.6.6A.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at ~~[the first refueling and at]~~ 10 year intervals is considered adequate to detect obstruction of the nozzles.

SR 3.6.6.9

(Continued)

C

BASES

assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves 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 and capable of potentially being mispositioned are in the correct position.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The required volume may be surveilled using an indicated level band of 50 to 88% for the Spray Additive Tank which corresponds to the LCO 3.6.7 minimum and maximum limits adjusted conservatively for instrument accuracy of ±0.3%. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and equipped with a low level alarm in the control room, so that there is high confidence that a ~~substantial change in~~ level below an acceptable value would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The ~~18~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~18~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(Continued)

BASES (Continued)

d. Corrosion of metals exposed to containment spray and Emergency Core Cooling System solutions.

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

Based on the conservative assumptions used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach 3.5 v/o about 616 days after the LOCA and 4.0 v/o about 2 days later if no recombiner was functioning (Ref. 3). Initiating the hydrogen recombiners when the primary containment hydrogen concentration reaches 3.5 v/o will maintain the hydrogen concentration in the primary containment below flammability limits.

The ~~hydrogen~~ recombiners are designed such that, with the conservatively calculated hydrogen generation rates discussed above, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 v/o (Ref. ~~4~~ 3). The Hydrogen Purge System is ~~similarly~~ designed and constructed such that ~~one of two redundant trains~~ it is Design Class I (for Quality and electrical power) but not redundant. As such, it is an adequate backup to the redundant hydrogen recombiners since it would be relied upon only in the event of a non-design basis condition.

The hydrogen recombiners satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii).

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst case single failure.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

(Continued)

[illegible]

BASES (Continued)

C.1

If the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status within the required Completion Time, 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. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.8.1

Performance of a system functional test for each hydrogen recombiner ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum power for approximately 2 minutes and power is verified to be ≥ 60 kW.

Operating experience has shown that these components usually pass the Surveillance when performed at the ~~18~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.8.2

This SR ensures there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failure involves loss of power, blockage of the internal flow, missile impact, etc.

A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The ~~18~~ 24 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.8.3

This SR, which is performed following the functional test of SR 3.6.8.1, requires performance of a resistance to ground test for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms.

The ~~18~~ 24 month Frequency for this Surveillance was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

(Continued)

DCL-97-106, LAR 97-09
ITS 3.7 ERRATA

(E)

3/4.7 PLANT SYSTEMS

3/4.7.1 ~~TURBINE CYCLE~~ MAIN STEAM SAFETY VALVES (MSSVs)

~~SAFETY VALVES~~

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE ~~with lift settings as specified in Table 3.7-2~~. 01-01-A

APPLICABILITY: MODES 1, 2 and 3.

ACTION*:

a. With one or more main steam line Code safety valves inoperable. ~~operation in MODES 1, 2 and 3 may proceed provided, that~~ within 4 hours, either the inoperable valve is restored to OPERABLE status or ~~the Power Range Neutron Flux High Trip Setpoint is reduced~~ reduce power per Table 3.7-1; otherwise, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours. 01-02-LS1 01-04-LS3

(new) # Reduce the Power Range High Neutron Flux reactor trip set points to less than or equal to the Maximum Allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs within 72 hours. Otherwise, be in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. 01-04-LS3

~~b. The provisions of Specification 3.0.4 are not applicable.~~ 01-05-M

(new) With one or more steam generators with less than two MSSVs OPERABLE, be in at least MODE 3 within the next 6 hours and in MODE 4 within 12 hours. 01-06-M

SURVEILLANCE REQUIREMENTS

4.7.1.1** ~~No additional requirements other than those required by Specification 4.0.5.~~ Verify each required MSSV lift setpoint per Table 3.7-2 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ± 1%. 01-07-A

* Separate Condition entry is allowed for each MSSV. 01-02-LS1

** Only required to be performed in MODES 1 and 2. 01-05-M

\# Only Applicable in MODE 1. 01-04-LS3

(B)

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER ~~RANGE NEUTRON FLUX HIGH SETPOINT~~ WITH INOPERABLE STEAM LINE SAFETY VALVES 01-04-LS3

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER ~~RANGE NEUTRON FLUX HIGH SETPOINT~~ ~~(~~PERCENT OF RATED THERMAL POWER~~)~~
1	~~87*~~ ~~≤ 92~~ 87
2	~~64*~~ ~~≤ 45~~ 47
3	~~42*~~ ~~≤ 27~~ 29

~~01-09-LS31~~
01-04-LS3

~~*Unless the Reactor Trip System breakers are in the open position.~~ 01-04-LS3

(C)

(2) Verifying that each ~~non-automatic~~ manual, power operated and automatic valve in the ~~pump~~ water flow path and both steam supplies to the steam turbine-driven pump that is not locked, sealed, or otherwise secured in position, is in its correct position. 02-07-M 02-09-A

~~(3) Verifying that each non-automatic valve in both steam supplies to the steam turbine-driven pump that is not locked, sealed, or otherwise secured in position, is in its correct position.~~ 02-09-A

b. * ~~At least once per 92 days on a STAGGERED TEST BASIS by:~~ Testing the steam turbine-driven pump and motor-driven pumps ~~pursuant to~~ in accordance with the frequency specified in ~~Specification 4.0.5*~~ the Inservice Testing Program. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the steam turbine-driven pump. 02-08-LS6 02-04-M

c. ** At least once each REFUELING INTERVAL ~~per 18 months~~ by verifying that each ~~auxiliary feedwater pump starts~~ ~~and~~ valve opens* as designed automatically upon receipt of ~~an~~ a simulated or actual ~~Auxiliary Feedwater Actuation test~~ signal. 02-11-A 02-12-TR1

(new)
* ** At least once each REFUELING INTERVAL ~~per 18 months~~ by verifying each AFW pump starts automatically on an actual or simulated actuation signal. 02-11-A 02-12-TR1 02-04-M

* Not required to be performed ~~f~~For the steam turbine-driven pump, ~~when~~ until 24 hours after the secondary steam supply pressure is greater than 650 psig. 02-14-M

** Not applicable in MODE 4 when steam generator is relied upon for heat removal 02-04-M

(D)

PLANT SYSTEMS

AUXILIARY FEEDWATER SOURCES - CONDENSATE STORAGE TANK (CST) AND FIRE WATER STORAGE TANK (FWST)

Add strike-out per 03-01-LG

LIMITING CONDITION FOR OPERATION

3.7.1.3 The Condensate Storage Tank (CST) level shall ~~have a usable volume of~~ be at least ~~164,678 gallons of water~~ 41.3% ~~with an open flow path to the Auxiliary Feedwater (AFW) pump suction,~~ and the Fire Water Storage Tank (FWST) level shall ~~have a usable volume of~~ be at least ~~57,922 gallons of water~~ 22.2% for one Unit operation and ~~115,844 gallons of water~~ 41.7% for two Unit operation ~~with a flow path capable of being aligned to the AFW pump suction~~. 03-01-LG

APPLICABILITY: MODES 1, 2 and 3~~.~~,
MODE 4 when steam generator is relied upon for heat removal. 02-04-M

ACTION:

a. With the CST ~~usable volume less than 164,678 gallons~~ level not within limit, or with the CST flow path not open to the AFW pumps suction, within four hours restore the required CST conditions; or, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN without reliance upon steam generator for heat removal within the following ~~6~~12 hours. 02-04-M 02-10-LS21

b. With an FWST ~~usable volume less than 57,922 gallons for one Unit operation and 115,844 gallons for two Unit operation~~ level not within limits, or with the FWST flow path not capable of being aligned to the AFW pump suction, within seven days restore the required FWST conditions; or, be in at least HOT STANDBY, within the next 6 hours and in HOT SHUTDOWN without reliance upon steam generator for heat removal within the following ~~6~~12 hours. 02-10-LS21 02-04-M

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The CST volume shall be demonstrated at least once per 12 hours by verifying the ~~usable volume~~ level is within its limits.

4.7.1.3.2 The FWST volume shall be demonstrated at least once per 12 hours by verifying the ~~usable volume~~ level is within its limits.

4.7.1.3.3 Verify the FWST is capable of being aligned to the Auxiliary Feedwater System by cycling each FWST valve in the flow path necessary for realignment through at least one full cycle once per quarter. 02-13-A

PLANT SYSTEMS

(C)

~~STEAM GENERATOR~~ 10% ATMOSPHERIC DUMP VALVES (ADVs)

LIMITING CONDITION FOR OPERATION

3.7.1.6 Four ~~steam generator~~ 10% ~~atmospheric dump valves~~ ~~(ADVs)~~ ~~with the associated block valves open and associated remote manual controls, including the backup air bottles.~~ lines shall be OPERABLE. 06-01-LG 06-09-A

APPLICABILITY: MODES 1, 2, and 3~~.~~, MODE 4 when steam generator is relied upon for heat removal. 02-04-M

ACTION: 06-02-LS14

a. With one less than the required number of 10% ADV~~s~~ lines OPERABLE, restore the inoperable ~~steam generator~~ 10% ADV line to OPERABLE status within 7 days (3.0.4 is not applicable); or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN without reliance upon steam generator for heat removal within the following ~~6~~12 hours. 02-10-LS21 02-04-M

b. With two less than the required numbered of 10% ADV~~s~~ lines OPERABLE, restore at least one of the inoperable steam generator 10% ADVs to OPERABLE status within 72 hours; or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN without reliance upon steam generator for heat removal within the following ~~6~~12 hours. 02-10-LS21 02-04-M

(new) With three or more ADV lines inoperable, restore at least two to an OPERABLE status within 24 hours or be in MODE 3 in 6 hours and be in MODE 4 without reliance upon steam generators for heat removal within 18 hours. 06-05-LS24 02-10-LS21 02-04-M

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each ~~steam generator~~ 10% ADV line~~s~~, ~~associated block valve and associated remote manual controls including the backup air bottles~~ shall be demonstrated OPERABLE: 06-01-LG

a. At least once per 24 hours by verifying that the backup air bottle for each steam generator 10% ADV has a pressure greater than or equal to 260 psig, and

b. ~~At least once per 31 days by verifying that the steam generator 10% ADV block valves are open, and~~ 06-06-LG

c. At least once each REFUELING INTERVAL ~~per 18 months~~ by verifying that all steam generator 10% ADVs will operate ~~using the remote manual controls and the backup air bottles.~~ 06-06-LG

(new) Verify one complete cycle of each 10% ADV block valve once per 18 months. 06-04-M

PLANT SYSTEMS

3/4.7.3 VITAL COMPONENT COOLING WATER ~~(CCW)~~ SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 At least two vital component cooling water loops shall be OPERABLE.

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

ACTION:* 08-02-A

With only one vital component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 At least two vital component cooling water loops shall be demonstrated OPERABLE:

a. ** At least once per 31 days ~~by~~ verify~~ing~~ that each CCW valve (manual, power-operated, or automatic) in the flow path servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position; and 08-04-A 08-08-A

each REFUELING INTERVAL

b. At least once ~~per 18 months~~; ~~by~~ verify~~ing~~ that each automatic valve in the flow path that is not locked, sealed or otherwise secured in position ~~servicing safety-related equipment~~ actuates to its correct position on ~~a Safety Injection or Phase "B" Isolation test~~ an actual or simulated actuation signal~~, as appropriate~~. 08-08-A 08-05-A

(new) Verify each CCW pump starts automatically on an actual or simulated actuation signal at least once ~~per 18 months~~. 08-06-TR1 08-07-M

each REFUELING INTERVAL

* Enter applicable conditions and required actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW. 08-02-A

** Isolation of CCW flow to individual components does not render the CCW system inoperable. 08-04-A

(C)

PLANT SYSTEMS

3/4.7.4 AUXILIARY SALTWATER SYSTEM (ASW)

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two auxiliary saltwater trains shall be OPERABLE.

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

ACTION:* 08-02-A

With only one auxiliary saltwater train OPERABLE, restore at least two trains to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 At least two auxiliary saltwater trains shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, or power-operated~~, or automatic~~) in the flow path servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position or that power and/or air supplies are available such that the valve would be capable of being placed in its correct position. 09-03-A 09-01-M

(new) Verify each ASW pump starts automatically on an actual or simulated actuation signal at least once ~~per 18 months~~. 09-04-M
each REFUELING INTERVAL

(new) Verify each ASW power operated valve in the flow path that is not locked, sealed or otherwise secured in position is capable of being placed in the correct position in accordance with the Inservice Testing Program. 09-06-M

* Enter applicable conditions and required actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by ASW/CCW. 08-02-A

b. At least once per 31 days by:

1) ~~Initiating flow through the HEPA Filter And Charcoal Adsorber System and~~ verifying that ~~either redundant set of booster and pressurization supply fans~~ the system operates for at least 10 continuous hours with the heaters operating, 10-06-LG

2) ~~Verifying that each Ventilation System redundant fan is aligned to receive electrical power from a separate OPERABLE vital bus, and~~ 10-16-LG

3) ~~Starting (unless already operating) each main supply fan, booster fan, and pressurization supply fan, and verifying that it operates for 1 hour.~~ 10-06-LG

(new) Perform required CRVS FILTER TESTING in accordance with the Ventilation Filter Testing Program (VFTP).

c. At least once per REFUELING INTERVAL or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by: 10-09-A 10-09-A

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in ANSI N510-1980, and the system flow rate is 2100 cfm ± 10%;

2) ~~Verifying a system flow rate of 2100 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.~~ 10-17-A

d. At least once per 18 months, or (1) after any structural maintenance on the charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, or (3) after 720 hours of charcoal adsorber operation, by verifying, ~~within 31 days after removal,~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30°C and at 70% R.H. for a methyl iodide penetration of less than 1%; 10-09-A 10-23-LS13 10-24-A

C

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per REFUELING INTERVAL ~~18 months~~ by: 10-08-A

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3.5 inches Water Gauge while operating the system at a flow rate of 2100 cfm ± 10%;

2) Verifying that on an actual or simulated actuation ~~a Phase "A" Isolation test~~ signal, the system automatically switches into the pressurization mode of operation. ~~with approximately 27% (determined by damper position) of the flow through the HEPA filters and charcoal adsorber banks;~~ 10-10-TR1 10-08-A

3) Verifying, on a STAGGERED TEST BASIS, that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during the pressurization mode of system operation; and 10-11-LS19

4) Verifying that the heaters dissipate 5 ± 1 kW when tested in accordance with ANSI N510-1980. 10-08-A

f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 or a DOP test aerosol while operating the system at a flow rate of 2100 cfm ± 10%; and 10-08-A

g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon test gas while operating the system at a flow rate of 2100 cfm ± 10%. 10-08-A

C

PLANT SYSTEMS

3/4.7.6 AUXILIARY BUILDING ~~SAFEGUARDS AIR FILTRATION~~ VENTILATION SYSTEM (ABVS)

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two Auxiliary Building ~~Safeguards Air Filtration~~Ventilation System exhaust trains with one common HEPA filter and charcoal adsorber bank and at least two supply and exhaust fans shall be OPERABLE. 12-10-M

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

ACTION:

a. With the HEPA filter and charcoal adsorber bank inoperable, restore the HEPA filter and charcoal adsorber bank to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With only one supply and one exhaust fan OPERABLE, restore at least two supply and two exhaust fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. 12-10-M

SURVEILLANCE REQUIREMENTS

4.7.6.1 Each Auxiliary Building ~~Safeguards Air Filtration~~Ventilation System train shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) ~~Initiating flow through the HEPA filter and charcoal adsorber bank and~~ verifying that the train operates for at least 10 continuous hours with the heaters operating. ~~and~~ 12-06-LG

~~2) Verifying that each exhaust fan is aligned to receive electrical power from a separate OPERABLE vital bus.~~ 12-06-LG

(new) Perform required ABVS filter system testing in accordance with the Ventilation Filter Testing Program (VFTP). 10-08-A

b. At least once per ~~18 months~~ REFUELING INTERVAL or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, by: 10-08-A

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in ANSI N510-1980, and the system flow rate is 73,500 cfm ± 10%;

2) Verifying a system flow rate of 73,500 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980. 10-17-A

c. At least once per 18 months, or (1) after any structural maintenance on the charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, or (3) after every 720 hours of charcoal adsorber operation, by verifying, ~~within 31 days after removal~~, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30°C and at 70% R.H. for a methyl iodide penetration of less than 6%; 10-09-A 10-23-LS13

d. At least once per REFUELING INTERVAL by: 10-08-A

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 3.7 inches Water Gauge while operating the system at a flow rate of 73,500 cfm ± 10%,

2) Verifying that each train actuates and that flow is established through the HEPA filter and charcoal adsorber bank on ~~a Safety Injection test~~ an actual or simulated actuation signal, and 12-10-M 12-04-TR1

3) Verifying that the heaters dissipate 50 ± 5 kW when tested in accordance with ANSI N510-1980. 10-08-A

4) Verifying that leakage through the Auxiliary Building Safeguards Air Filtration System Dampers M2A and M2B is less than or equal to 5 cfm when subjected to a Constant Pressure or Pressure Decay Leak Rate Test in accordance with ASME N510-1989. The test pressure for the leak rate test shall be based on a maximum operating pressure as defined in ASME N510-1989, of 8 inches water gauge.

A note is added per WOG-83 Rev.1 that states that the flux trip reduction ACTION is only Applicable in MODE 1.

E

CHANGE NUMBER	NSHC	DESCRIPTION
		the CTS requirement to reset the power range neutron flux-high trip setpoints based on the number of MSSVs inoperable to a maximum allowable power determined in accordance with calculations or analysis to account for Westinghouse NSAL 94-001 and NRC Information Notice 94-60. However, the Completion Time of 72 hours proposed by WOG-83 has been retained and is justified based on the low probability of an event occurring during this time and the need to provide sufficient time to reset the channels in an orderly manner without inducing a transient due to human error. Retention of the CTS requirement for resetting the reactor trip setpoints is acceptable because this requirement is more conservative than the ACTIONS specified by either the ISTS or WOG-83, as revised. [ ]
01-05	M	The exception to TS 3.0.4 is no longer needed due to the note associated with the revised surveillance. The exception was allowed to TS 3.0.4 due to the fact that the applicable MODES must be entered in order to perform the required surveillance (if the MSSVs are tested in place) and to allow Mode changes to be made if the applicable action was met. In the CTS, MODE 1, 2, or 3 could be entered. In NUREG-1431, the surveillance is modified by a more restrictive note that specifies that the surveillance need only be current prior to reaching MODE 2. The surveillance note still allows MODE changes into the MODE of Applicability of the LCO, i.e., MODE 3 for testing purposes.
01-06	M	The new ACTION adds an explicit requirement to be in MODE 3 in 6 hours and MODE 4 in 12 hours if any steam generator (SG) loop has less than two MSSVs OPERABLE. NUREG-1431 requires that the plant only be placed in a MODE where the specification is no longer applicable, which in this case would be MODE 4. The CTS would require the plant to enter TS 3.0.3 because operation with less than two MSSVs OPERABLE per SG is an undefined condition, and thus not permitted. Therefore, the new ACTION eliminates the one hour allowed for action via TS 3.0.3. This requirement is more restrictive with the loss of the one hour for ACTIONs required by TS 3.0.3.
01-07	A	The CTS SR is revised to specifically reference the In-service Testing (IST) Program developed per TS 4.0.5 and contained in the Administrative Section of the ITS. The surveillance directly references Table 3.7-2 for lift points and incorporates the requirement that the MSSV as-left liftpoints to be within +/- 1 percent of the nominal setpoint.
01-08		Not Used.

B ~~&~~ E

CHANGE NUMBER	NSHC	DESCRIPTION
01-09	LS31	Not Used. ~~The maximum power range neutron high flux trip setpoints required for one or more inoperable MSSVs are revised in accordance with the recommendations of Westinghouse NSAL 94-001, dated January 20, 1994, and a specific calculation and analysis to support the proposed revision. Since the issuance of NSAL 94-001, administrative controls have been in place to require a reactor power and high neutron flux trip setpoint reduction to be consistent with those values determined by NSAL 94-001 or as required by the CTS if they were more conservative. Since the trip setting of the CTS Table 3.7-1 is revised upward for one inoperable MSSV for incorporation into the ITS, this change has been classified as less restrictive. The settings for two or three inoperable MSSVs are more restrictive than the GTS or NUREG-1431 as modified by WOG-83.~~
01-10	LG	The note on Table 3.7-2 stating that the set pressures of the MSSVs shall correspond to the ambient conditions of the valve at normal operating temperatures is moved to the Bases of ITS SR 3.7.1.1. This change is acceptable because it removes details from the TS that are not required to protect the health and safety of the public while retaining the basic limiting condition for operation.
01-11	LG	The MSSV ~~line~~ orifice size is moved from Table 3.7-2 to a licensee-controlled document. This is design information that is not required in the ITS for operating or OPERABILITY concerns.
01-12	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
02-01	LG	The descriptive material related to the definition of an auxiliary feedwater (AFW) train is deleted from the LCO and moved to the Bases. This change is acceptable because it removes details from the TS that are not required to protect the health and safety of the public while retaining the basic limiting condition for operation.
02-02	LS5	The ACTION specifies the requirements for allowed outage time (AOT) should one of the steam supplies to the turbine driven auxiliary feedwater (TDAFW) pump become inoperable. A previous interpretation required that the TDAFW pump be declared inoperable and the ACTION statement for one inoperable pump be entered. This revision is a relaxation of the CTS requirements.
02-03	M	The ACTIONs are modified to require restoration of the systems to meet the LCO within 10 days of discovery of failure to meet the LCO. This new requirement is intended to prevent multiple overlapping ACTION entries such that the intended AOT is exceeded. This proposed change is acceptable because it results in more stringent TS requirements that are both appropriate and consistent with NUREG-1431.

(8)

CHANGE NUMBER	NSHC	DESCRIPTION
06-02	LS14	This change adds an exception to LCO 3.0.4 for the 7 day ACTION to restore the atmospheric [dump] valve OPERABILITY. This change allows the plant to change MODES if one atmospheric [dump] valve is found inoperable while in MODE 2 or 3. Allowing MODE transition with an inoperable atmospheric [dump] valve does not significantly increase the risk since the remaining valves are OPERABLE.
06-03	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B)
06-04	M	A surveillance is added that requires the manual cycling of the atmospheric [dump] valve [block] valves [every 18 months]. This proposed change is acceptable because it results in more stringent TS requirements that are both appropriate and consistent with NUREG-1431.
06-05	LS24	This change adds a new ACTION for three or more inoperable atmospheric [dump] valves that requires action within 24 hours. The CTS would require entry into TS 3.0.3 for three inoperable atmospheric [dump] valves. However, NUREG-1431 recognizes the low probability of an accident requiring the atmospheric [dump] valves, thus permitting this configuration.
06-06	LG	This change moves the requirements for the surveillances to the Bases. The details of the specific test requirements such as those dealing with the "remote manual controls" are not included in the STS, but are included in the Bases section for the SR. [Verification that the block valves to the 10% ADVs are open is moved to the Bases as a procedural surveillance.] This change is acceptable because it removes details from the TS that are not required to protect the heath and safety of the public while retaining he basic limiting condition for operation. t
06-07	LS25	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
06-08	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
06-09	A	This change revises the LCO to refer to the atmospheric [dump] valve lines versus atmospheric [dump] valves.
06-10	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
07-01	A	The LCO is changed from the OPERABILITY of each main feedwater line to the OPERABILITY of four feedwater flow path isolation valves [and associated bypasses] [and the requirement to have the valves closed or isolated if not OPERABLE is moved to Applicability. The ACTIONS are also revised to compensate for the line requirement deletion].
07-02	LS37	The LCO Applicability is revised to exclude [MFIVs or FRVs] that are closed [and de-activated or isolated by a closed manual valve]. NUREG-1431 recognizes that if one or more [MFIVs or FRVs] or associated bypass valves are closed [and de-activated] and verified closed, their safety function is being fulfilled and there is no need to enter the ACTION statement.

C

CHANGE NUMBER	NSHC	DESCRIPTION
08-02	A	A Note is added to the ACTION that references a potential interaction with ITS 3.4.6 dealing with OPERABILITY of the RHR system in MODE 4. The Note requires that the applicable TS be entered for the RHR train made inoperable by the inoperable [component cooling water (CCW) or auxiliary saltwater (ASW)] system. The ACTIONS of the referenced TS (RCS Loops-MODE 4) require more immediate action than are required by the [CCW or ASW] ACTIONS.
08-03	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
08-04	A	A Note is added to the [CCW] surveillance that clarifies that the system is not made inoperable by the isolation of individual components. This change is in accordance with NUREG-1431, and provides clarification only.
08-05	A	The surveillance is modified to clarify that valves that are locked, sealed, or otherwise secured in their correct position are not required to be tested. This change is in accordance with NUREG-1431, and provides clarification only.
08-06	TR1	The SR is revised to allow credit for an actual actuation, if one occurs, to satisfy the SRs. The identification of the signal is moved to the Bases.
08-07	M	A new surveillance is added that requires verifying that each CCW pump starts automatically on an actual or simulated signal actuation at least once per ~~18~~ 24 months.
08-08	A	The surveillances are revised to clarify that only verification of the correct position of valves in the flow path is required.
08-09		Not used.
08-10	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
09-01	M	This change revises the existing surveillance to verify that a motive source is available that would permit the required ASW valves to be repositioned. This change is consistent with the intent of NUREG-1431.
09-02	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
09-03	A	The surveillance is reworded and the requirement to verify the position of automatic valves is deleted since the ASW system has no automatic valves.
09-04	M	A new surveillance is added that requires verifying that each ASW pump starts automatically on an actual or simulated actuation signal at least once per ~~18~~ 24 months.
09-05	TR1	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
09-06	M	A new surveillance is added that requires verifying that each valve in the flow path that is not locked, sealed, or otherwise secured in position is capable of being placed in the correct position in accordance with the Inservice Testing (IST) Program.

(C)

CHANGE NUMBER	NSHC	DESCRIPTION
10-11	LS19	The Frequency of the surveillance requiring verification of the CR ventilation system capability to maintain a positive pressure is relaxed to ~~18~~ months on a STB, consistent with NUREG-1431. The new Frequency requires one of the two trains to be tested every ~~18~~ months instead of both trains every ~~18~~ months. The most likely cause of a failure to achieve the required pressure is a failure of the ventilation pressure boundary. Thus, when one train successfully demonstrates the ability to maintain the pressure, in all likelihood the other train will also. This results in less testing of the CR ventilation system than is required by the CTS. [24]
10-12	LS32	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-13	LG	The footnotes indicating that CR ventilation system is common to both units, and that the system may be considered OPERABLE with no chlorine monitors if no bulk chlorine gas is stored within the SITE BOUNDARY, are moved to the Bases.
10-14	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-15	LG	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-16	LG	The requirement to have an available emergency power source for the CR ventilation system is moved to the Bases. This change is consistent with the Bases of NUREG-1431 for ventilation systems required during fuel movement.
10-17	A	The SR to measure ventilation system flow rate is not identified as a separate SR in the ITS because it is verified during the other in-place filter tests (see ITS 5.5.11).
10-18	LS36	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-19	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-20	LS39	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
10-21	LS38	The ACTION to immediately suspend all operations involving CORE ALTERATIONS and movement of irradiated fuel assemblies when both trains of CR ventilation are inoperable in MODES 5 and 6 and during movement of irradiated fuel assemblies is deleted consistent with NUREG-1431. This change is acceptable because the immediate suspension of CORE ALTERATIONS and movement of irradiated fuel provides adequate protection from a release of radioactivity. Boron dilution events leading to criticality are not postulated as these events are prevented from occurring.
10-22	M	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
01-07 A	The CTS SR is revised to specifically reference the IST Program. The surveillance directly references Table 3.7-2 for lift points and incorporates the footnote from the table requiring the MSSV as left liftpoints to be within +/- 1% of the nominal setpoint.	Yes	Yes	Yes	Yes
01-08	Not used.	NA	NA	NA	NA
01-09 LS31	~~This DCPP specific change revises the maximum power range neutron high flux trip setpoints required for one or more inoperable MSSVs in accordance with the recommendations of Westinghouse NSAL 94-001, dated January 20,1994, and specific analysis and calculations performed to confirm the conclusions of the Westinghouse NSAL.~~ Not Used.	~~Yes; LAR 97-08 submitted justifying revised high flux trip set points for inoperable MSSVs.~~ NA	~~No; refer to 01-04-LS3.~~ NA	~~No; refer to 01-04-LS3.~~ NA	~~No; refer to 01-04-LS3.~~ NA
01-10 LG	The Note on Table 3.7-2 stating that the set pressures shall correspond to the ambient conditions of the valve at normal operating temperatures is moved to the Bases of ITS SR 3.7.1.1.	Yes	Yes	Yes	Yes
01-11 LG	The MSSV ~~line~~ orifice size is moved to a licensee-controlled document.	Yes; moved to FSAR.	Yes; moved to FSAR.	Yes; moved to USAR.	Yes; moved to FSAR.
01-12 A	The proposed change would require that the plant be placed in HOT SHUTDOWN within 12 hours instead of COLD SHUTDOWN within 36 hours.	No; already part of CTS.	No; part of CTS.	Yes	Yes
02-01 LG	The descriptive material, definition of an AFW train, in the LCO is moved to the Bases.	Yes	Yes	Yes	Yes

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TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
08-02 A	A Note is added to ACTION that references potential interaction with ITS 3.4.6 for RHR MODE 4 operability.	Yes	Yes	Yes	Yes
08-03 LG	The requirement to perform the 18 month surveillance "during shutdown" would be moved to the Bases.	No, not in CTS.	No, not in CTS.	Yes	Yes
08-04 A	A Note is added that clarifies [CCW] operability.	Yes	Yes	Yes	Yes
08-05 A	Surveillance is modified to exclude valves that are locked, sealed, or otherwise secured in their correct position.	Yes	Yes	Yes	Yes
08-06 TR1	The SR is revised to allow credit for an actual actuation and moves signal specifics to the Bases.	Yes	Yes	Yes	Yes
08-07 M	A new surveillance specific to DCPP is added that requires verifying that each CCW pump starts automatically on an actual or simulated signal actuation at least once per ~~18~~ 24 months.	Yes	No	No	No
08-08 A	Surveillance is modified to only be applicable to flow path valves.	Yes	Yes	Yes	Yes
08-09	Not Used.	NA	NA	NA	NA
08-10 A	The Callaway specific Note applicable to cycle 1 surveillance requirements is no longer needed.	No	No	No	Yes
09-01 M	A DCPP existing surveillance is revised that requires verifying that a motive source is available that would allow the required valves to be repositioned.	Yes	No	No	No

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
09-02 A	A Note is added that requires entry into applicable LCOs if an inoperable [ASW] system makes the affected equipment inoperable.	No, ASW only supplies CCW heat exchangers.	Yes	Yes	Yes
09-03 A	The DCPP specific surveillance is reworded and the requirement to verify the position of automatic valves is deleted since the ASW system has no automatic valves.	Yes	No	No	No
09-04 M	A new surveillance specific to DCPP is added that requires verifying that each ASW pump starts automatically on an actual or simulated actuation signal at least once per ~~18~~ 24 months.	Yes	No	No	No
09-05 TR1	The SR is revised to allow credit for an actual actuation and moves signal specifics to the Bases.	No, refer to 09-04-M. This requirement did not previously exist.	Yes	Yes	Yes
09-06 M	A new surveillance specific to DCPP is added that requires verifying that each valve in the flow path that is not locked, sealed, or otherwise secured in position is capable of being placed in the correct position in accordance with the IST Program.	Yes, also refer to change 09-01-M.	No	No	No
09-07 A	A Note is added to the [ASW] surveillance that clarifies system OPERABILITY requirements. Isolation of [ASW] flow to individual components does not render the system inoperable.	No, ASW only supplies CCW heat exchangers.	Yes	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
10-08 A	The description of the ventilation filter specific testing requirements are moved to the VFTP, as defined in the Administrative Controls of the ITS, or deleted as being duplicated in the applicable RGs or Standards. A SR is added that requires [CR and Auxiliary Building ventilation system] filter testing in accordance with the VFTP.	Yes	Yes	Yes	Yes
10-09 LS27	The ACTION for an OPERABLE ventilation train not being capable of being supplied from an emergency power source is deleted.	No, refer to change 10-16-LG.	No, not in CTS.	Yes	Yes
10-10 TR1	The SR is revised to allow credit for an actual actuation and moves signal specifics to the Bases.	Yes	Yes	Yes	Yes
10-11 LS19	Frequency of the surveillance requiring verification of the control room ventilation system capability to maintain a positive pressure in the CR is relaxed to ~~18~~ [24] months on a STB.	Yes	Yes	Yes	Yes
10-12 LS32	Deletes the STB for the 31 day testing.	No, CTS surveillance is not STB.	No, not in CTS.	Yes	Yes
10-13 LG	The DCPP specific footnotes indicating the control room ventilation system is common to both units and that the system may be considered OPERABLE with no chlorine monitors if no bulk chlorine gas is stored within the SITE BOUNDARY, are moved to the Bases.	Yes	No	No	No

(E)

NO SIGNIFICANT HAZARDS CONSIDERATION

(continued)

IV. Specific No Significant Hazards Considerations - "LS" (continued)

Recurring No Significant Hazards Considerations - "TR"

(E)

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS3
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS allow continued operation with inoperable MSSVs if the power range neutron flux high reactor trip setpoints are reduced. The amount of reduction in the trip setpoint is dependent on the total number of inoperable MSSVs per SG and is intended to compensate for the lost relief capacity (heat removal capability and thus overpressure protection) should a transient requiring their operation occur. In the proposed specification, the CTS requirement to reduce the power range high neutron flux reactor trip setpoint is retained; however, the time to complete resetting the trip setpoints would be changed from four to 72 hours.

[in MODE 1]

The CTS require that, if the MSSV cannot be restored to an OPERABLE status within four hours, the power range high neutron flux reactor trip setpoints must be reset in the same 4-hour period. NUREG-1431 requires that the reactor power be reduced in four hours if the MSSV cannot be returned to an OPERABLE status; however, NUREG-1431 would not require resetting the power range neutron flux high setpoints. The Westinghouse Owners Group (WOG) has proposed changes to NUREG-1431 (traveler WOG-83, as revised through draft Rev. 1) that: 1) propose that the completion time for resetting the power range neutron flux high trip setpoint to compensate for a positive MTC or a control rod withdrawal event at partial reactor power to be 72 hours, 2) specifies that power level reductions be per the Westinghouse Nuclear Safety Advisory Letter, NSAL 94-01 and 3) deletes the Maximum Allowable % RTP for 5 MSSVs OPERABLE. However, pending approval of draft Rev. 1 of WOG-83, the changes proposed in the traveler have been modified to retain the current TS requirement to reset the power range neutron flux-high trip setpoints based on the number of MSSVs inoperable to a maximum allowable power determined in accordance with calculations or analysis to account for Westinghouse NSAL 94-001. The allowed Completion Time to reduce the Power Range Neutron Flux trip setpoints is reasonable based on operating experience to accomplish the required ACTIONs in an orderly manner. The power levels specified per NSAL 94-001 are based on a conservative algorithm developed by Westinghouse to bound the required relief capacity.

The above changes are consistent with NUREG-1431 as revised by WOG-83 and NSAL-94-001. This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> 1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
> 2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
> 3. *Involve a significant reduction in a margin of safety."*

C

IV. SPECIFIC NO SIGNIFICANT HAZARD CONSIDERATIONS

NSHC LS19
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The CTS require that the CR [emergency filtration (ventilation) system (CREFS)] trains be tested at least once every ~~18~~ months by verifying that the system maintains the control room at a positive pressure of greater than or equal to [1/8 inches] of water gauge relative to the outside atmosphere during the pressurization MODE of operation. This testing is currently performed on both [CREFS] trains as required by the CTS. NUREG-1431 would revise the Frequency to at least once every ~~18~~ months on a STB, which would require testing only one train every ~~18~~ months. This revised testing Frequency is consistent with NUREG-0800, Section 6.4 for proving the [CR] pressure boundary integrity. The test will still evaluate the integrity of the CR structure and the ability of the [CR ventilation system] to maintain a positive pressure with respect to the outside atmosphere and adjacent areas every ~~18~~ months. The [CR ventilation system] OPERABILITY verification conducted every 31 days is not revised and will verify OPERABILITY of the system components. [24]

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> *(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> *(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> *(3) Involve a significant reduction in a margin of safety."*

The following evaluation is provided for the NSHC.

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

   The proposed change does not alter the unit configuration or operation or the function of any safety system. Consequently, the change does not increase the probability of an accident as defined in the FSAR Update. Revising the testing Frequency to verify the CR pressure boundary consistent with NRC guidance does not effect the analyzed accident, its probability, or its consequences.

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

(E)

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS31
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

Delete this LS. Not used

The CTS allow continued operation with inoperable MSSVs if the power range neutron flux high trip setpoints are reduced. The amount of reduction in the trip setpoint is dependent on the total number of inoperable MSSVs and is intended to compensate for the lost relieving capacity (heat removal capability) should a transient requiring their operation occur. On January 20, 1994, Westinghouse issued NSAL 94-001, which stated that the reductions in the power range neutron flux high trip setpoint required by TS Table 3.7-1 may not be bounding for the LOL/TT event since the power range neutron flux high trip setpoint specified may not be low enough to preclude secondary system overpressurization. In the proposed specification, the requirement to reduce the power range neutron flux high trip setpoint is revised such that the required setpoints satisfy the requirements of NSAL 94-001 or specific unit safety analyses. The reduction in trip setting prevents a power increase above those settings should the unit be operating with a positive MTC. The MSSVs are set to protect the secondary system against overpressurization in accordance with ASME codes and mitigate the consequences of anticipated operational transients. This change has been identified as less restrictive since the applicable set points have been revised consistent with NSAL 94-001 as opposed to NUREG-1431.

A unit may continue to operate with up to three MSSVs inoperable per steam line provided the power range neutron flux high trip setpoint is reduced as specified. Currently administrative controls are used to assure that the neutron high flux trip settings are reduced to appropriate levels per NSAL 94-001 should entry into the be required.

Upon receipt of Westinghouse NSAL 94-001, reanalysis of the LOL/TT event was performed. An analysis was performed for one MSSV inoperable on each SG that verified the plant could continue to operate at 100 percent rated thermal power. The analysis assumed worst case assumptions and that the lowest set MSSVs (the first to open during a pressure transient) were all inoperable. The analysis verified that the SG pressure would not reach 110 percent of design following a LOL/turbine trip transient. For two or three MSSVs inoperable on any SG, the algorithm recommended in Westinghouse NSAL 94-001, was used to calculate the power range neutron flux high trip setpoints. The algorithm recommended by Westinghouse is based on extremely conservative assumptions to determine the power level/steam flow that can be handled by the remaining OPERABLE MSSVs, i.e., that a reactor trip does not occur and that feedwater is unavailable. The calculation is documented and verified and specifies where the power range neutron flux high trip setpoints must be set to meet the required analysis. The required setpoints specified in Table 3.7-1 incorporate the uncertainties of the neutron flux measurement and the heat balance measurements as recommended by Westinghouse.

The NSHCs involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> *(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or*

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main feedwater regulating valve (MFRV) or associated MFRV bypass valve

, MFRVs or MFRV bypass valves

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS33
10 CFR 50.92 EVALUATION
FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

, 2 3

The proposed change would revise the ACTION for MODES 1 and 2, which currently apply to one inoperable [main feedwater isolation valve (MFIV)] to apply to one or more [MFIVs]. This is less restrictive than the current requirements which apply to only one inoperable valve.

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

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> *1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> *2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> *3. Involve a significant reduction in a margin of safety."*

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

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

MFRV, or MFRV bypass valve

The proposed change would have an insignificant effect on the probability of occurrence of an accident because the number of inoperable [MFIVs] would not affect any accident initiators. Therefore, the probability of an accident would not be significantly increased.

The operability of [MFIVs] could have an effect on the consequences of accidents that take credit for [MFIV] closure. However, these accidents are very low probability events that are not expected to occur during the lifetime of the unit. Nevertheless, should one of these low probability events occur, the accident analyses assume that various failures of equipment occur, and the failure of an [MFIV] is considered in these failure assumptions. Furthermore, other equipment would be expected to function to backup the [MFIV] function. These include feedwater check valves that prevent backflow through the feedwater lines, flow control valves that direct AFW flow away from a broken feedwater line, and feedwater control valves and feedwater pump trip circuits that can terminate feedwater flow. Effective decay heat removal from the unit can be accomplished with only one SG. These factors tend to mitigate the consequences of having more than one [MFIV] inoperable if an accident should occur.

E

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS33
(continued)

Therefore, the proposed change would have no significant effect on the probability or consequences of any previously analyzed accidents.

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

The proposed change involves the number of inoperable [MFIVs]. As noted above, there is other equipment available to backup the [MFIVs] should an accident occur. This equipment assures that an accident sequence would proceed as expected and analyzed in the unit safety analysis. Although some of the backup components are not in TS, they are designed to high standards and are periodically tested to assure operability. Therefore, the proposed change would not create the possibility of a new or different kind of accident.

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

The change involves the operability of equipment used to mitigate postulated accidents. As noted in the evaluation of Criterion 1 above, there is backup equipment available in the design to assist in performing the [MFIV] function. This equipment is expected to operate and would perform the [MFIV] functions in sufficient time to avoid a significant reduction in any margin of safety. Therefore, the proposed change would have no significant adverse effect on margins of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above evaluation, it is concluded that the activities associated with NSHC "LS33" resulting from the conversion to the ISTS format satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a NSHC finding is justified.

MFRV or MFRV bypass valve

MFRVs or MFRV bypass valves

3.7-01

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
~~Applicable~~ Maximum Allowable Power ~~in Percent of RATED THERMAL POWER~~

~~MINIMUM~~ NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR ~~REQUIRED OPERABLE~~	MAXIMUM ALLOWABLE ~~(%RTP)~~
	~~B-PS~~
~~5~~	~~≤ 109~~
4	~~≤ [65]~~ 87
3	~~≤ [46]~~ 47
2	~~≤ [28]~~ 29

For INFORMATION ONLY

** CORRECT AS IS **

(clean copy ITS + Enclosure 2 wrong)

(E)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.3.1 Verify the closure time of each MFIV is ≤ 60 seconds~~. MFRV [, and associated bypass valve] is ≤ [7] seconds on an actual or simulated actuation signal~~.	3.7-3 In accordance with the ~~[~~Inservice Testing Program ~~or [18] months]~~ B-PS 3.7-56
SR 3.7.3.2 Verify the closure time of each MFRV and associated bypass valve is ≤ 7 seconds.	At each COLD SHUTDOWN but not more frequently than once per 92 days. 3.7-3
SR 3.7.3.3 Verify each MFIV actuates to the isolation position on an actual or simulated actuation signal.	18 months 3.7-56

, MFRV and associated bypass valve

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each ADV.	~~18~~ months 24 B
SR 3.7.4.2	Verify one complete cycle of each ADV block valve.	18 months B
SR 3.7.4.3	Verify that the backup air bottle for each ADV has a pressure ≥260 psig.	24 hours 3.7-50

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CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Required Action and associated Completion Time for Condition F or G not met	H.1 Be in MODE 3 AND H.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours 3.7-9 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each AFW manual, power operated, and automatic valve in each water flow path, ~~[~~and in both steam supply flow paths to the steam turbine driven pump,~~]~~ that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days B
SR 3.7.5.2	-------------------NOTE------------------- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ ~~[1000]~~ 650 psig in the steam generator. --- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Test Program 3.7-29 B B-PS ~~[31] days on a STAGGERED TEST BASIS~~
SR 3.7.5.3	-------------------NOTE------------------- Not applicable in MODE 4 when steam generator is relied upon for heat removal. --- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	B ~~18~~ months 24

C

Surveillance Requirements (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.4	-------------------NOTES------------------- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ ~~[1000]~~ 650 psig in the steam generator. 2. Not applicable in MODE 4 when generator is relied upon for heat removal. --- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	B B-PS ~~18~~ 24 months B
SR 3.7.5.5	Not Used ~~Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.~~	B-PS ~~Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for > 30 days~~
SR 3.7.5.6	Verify the FWST is capable of being aligned to the AFW system by cycling each FWST valve in the flow path necessary for realignment through at least one full cycle.	92 days 3.7-9

C

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	-----------------NOTE---------------------- Isolation of CCW flow to individual components does not render the CCW System inoperable -- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each CCW automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~18~~ 24 months B
SR 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	~~18~~ 24 months B

C

Surveillance Requirements (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	~~NOTE~~ ~~Isolation of SWS flow to individual components does not render the SWS inoperable.~~ Verify each ~~SWS~~ ASW manual~~,~~ and power operated, ~~and automatic~~ valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position or that a motive force is available such that the valve would be capable of being placed in the correct position.	3.7-14 PS 31 days 3.7-15
SR 3.7.8.2	Verify each ~~SWS automatic~~ ASW power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, ~~actuates to~~can be moved to the correct position. ~~on an actual or simulated actuation signal~~	~~[18] months~~ B-PS PS 3.7-16 In accordance with the Inservice Test Program.
SR 3.7.8.3	Verify each ~~SWS~~ ASW pump starts automatically on an actual or simulated actuation signal.	~~18~~ 24 months B PS

(E)

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~One or more cooling towers with one cooling tower fan~~ With the UHS inoperable temperature > 64°F	A.1 ~~Restore cooling tower fan(s) to OPERABLE status.~~ Place a second CCW heat exchanger in service.	~~7 days~~ 8 hours B 3.7-17
B. Required Action and associated Completion Time of Condition A not met. ~~OR~~ ~~UHS inoperable [for reasons other than Condition A].~~	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours B 36 hours 3.7-17

C

SURVEILLANCE		FREQUENCY
SR 3.7.10.3	Verify each ~~CREFS~~ CRVS train ~~actuates~~ automatically switches into the pressurization mode of operation on an actual or simulated actuation signal.	~~18~~ 24 months B PS 3.7-33
SR 3.7.10.4	Verify one ~~CREFS~~ CRVS train can maintain a positive pressure of ≥ 0.125 inches water gauge, relative to the ~~adjacent [turbine building]~~ outside atmosphere during the pressurization mode of operation ~~at a makeup flow rate of ≤ [3000] cfm~~.	~~18~~ 24 months on a STAGGERED TEST BASIS 3.7-33 B P

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.12.1	Operate each ~~ECCS PREACS~~ ABVS train for ≥ 15 minutes, and one train for ≥ 10 continuous hours with the heaters operating ~~or (for systems without heaters) ≥ 15 minutes]~~.	31 days PS 3.7-21
SR 3.7.12.2	Perform required ~~ECCS PREACS~~ ABVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accord ance with the VFTP PS B
SR 3.7.12.3	Verify each ~~ECCS PREACS~~ ABVS train actuates on an actual or simulated actuation signal and the system realigns to exhaust through the common HEPA filter and charcoal adsorber.	~~18~~ 24 months PS B 3.7-22
SR 3.7.12.4	NOT USED ~~Verify one ECCS PREACS train can maintain a pressure ≤ [0.125] inches water gauge relative to atmospheric pressure during the [post accident] mode of operation at a flow rate of ≤ [3000] cfm.~~	~~[18] m ont hs on a STAGGERED TEST BASIS~~ 3.7-23
SR 3.7.12.5	NOT USED ~~Verify each ECCS PREACS filter bypass damper can be closed.~~	~~[18] m ont hs~~ 3.7-24
SR 3.7.12.6	Verifying that leakage through the ABVS Dampers M2A and M2B is less than or equal to 5 cfm when subjected to a Constant Pressure or Pressure Decay Leak Rate Test in accordance with ASME N510-1989. The test pressure for the leak rate test shall be based on a maximum operating pressure as defined in ASME N510-1989, of 8 inches water gauge.	~~18~~ 24 mon ths 3.7-18 PS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C. Required Action and associated Completion Time [of Condition A] not met during movement of irradiated fuel assemblies in the fuel building.~~	~~C.1 Place OPERABLE FBACS train in operation.~~ ~~OR~~ ~~C.2 Suspend movement of irradiated fuel assemblies in the fuel building.~~	~~Immediately~~ ~~Imme diat ely~~ 3.7-43
~~B~~D. Two ~~FBACS~~ FHBVS trains inoperable during movement of irradiated fuel assemblies in the fuel building.	~~B~~D.1 Suspend movement of irradiated fuel assemblies in the fuel handling building.	Immediately PS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each ~~FBACS~~ FHBVS train for ~~[≥ 10 continuous hours with the heaters operating or (for systems without heaters)~~ ≥ 15 minutes~~]~~.	31 days PS B-PS
SR 3.7.13.2	Perform required ~~FBACS~~ FHBVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP B PS
SR 3.7.13.3	Verify each ~~FBACS~~ FHBVS train actuates on an actual or simulated actuation signal.	~~18~~ 24 months B PS
SR 3.7.13.4	Verify one ~~FBACS~~ FHBVS train can maintain a pressure ≤ 0.125 inches water gauge with respect to atmospheric pressure during the post accident mode of operation ~~at a flow rate ≤ [20,000] cfm.~~	~~18~~ 24 months on a STAGGERED TEST BASIS 3.7-49 B PS

ⓔ

3.7-51

3.7 PLANT SYSTEMS

3.7.17 Spent Fuel Assembly Storage

LCO 3.7.17.1 The combination of initial enrichment, initial B-10 content, burnup, and storage pattern of each spent fuel assembly stored in Region 1 shall be:

a. The initial enrichment is 4.5 weight percent U-235 or less; or

b. The initial enrichment is from 4.5 up to a maximum of 5.0 weight percent U-235, and any of the following conditions are met:

1) The combination of initial enrichment and cumulative burnup of the assemblies is within the acceptable area of Figure 3.7.17-1, or

2) The assemblies initially contained a minimum of a nominal 36 mg/in. per assembly of the isotope B-10 integrated in the fuel rods, or

3) The assemblies are put in a checkerboard pattern with any of the following:

a) water cells, or

b) assemblies that initially contained a minimum of a nominal 72 mg/in. per assembly of the isotope B-10 integrated in the fuel rods, or

c) partially irradiated fuel of at least 8000 MWD/MTU cumulative burnup; or

4) The assemblies are put into a pattern with alternate rows of fuel assemblies and water cells.

APPLICABILITY: Whenever any fuel assembly is stored in Region 1 of the spent fuel pool.

(B)

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

APPLICABILITY

In ~~MODE 1 above [40] % RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. Below [40%] in~~ MODES 1, 2, and 3, ~~only two~~ ~~[~~ five ~~]~~ MSSVs per steam generator are required to be OPERABLE to limit secondary pressure.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 ~~[~~and A.2~~]~~

With one or more MSSVs inoperable, ~~reduce power~~action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements ~~for the applicable THERMAL POWER~~.

Continued operation with less than all ~~[~~five~~]~~ MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is ~~proportionally~~ limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER ~~[~~ and the Power Range Neutron Flux trip setpoint~~]~~ so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. ~~For example, if one MSSV is inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 20%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced by at least 20%. This is accomplished by reducing THERMAL POWER by at least 20%, which conservatively limits the energy transfer to all steam generators to approximately 80% of total capacity, consistent with the relief capacity of the most limiting steam generator.~~ ~~If one MSSV is inoperable on a SG, calculation N-114 (Ref. 8) demonstrates via RETRAN analysis that the secondary system pressure peak resulting from the limiting AOO is < 110% of design. The transient is terminated by a reactor trip, either high pressurizer pressure or OTDT, and the MSSVs maintain steam pressure below 110% at design.~~

Insert B

When a MSSV(s) is inoperable, the power must be reduced in 4 hours (required action A.1) to a value less than or equal to the value specified in table 3.7.1-1, corresponding to the number of OPERABLE MSSVs regardless of the value of the MTC.

INSERT B

(B)

The Reactor Trip Setpoint reductions applied in TS Table 3.7.1-1 are derived on the following bases:

One MSSV Inoperable

The limiting FSAR Condition II accident for overpressure concerns is a loss of external load/turbine trip. The event is analyzed with the RETRAN-02 computer program to demonstrate the adequacy of the MSSVs to maintain the main steam system lower than 1210 psia, or 110% of the 1085 psig SG design pressure.

In a PG&E calculation, the transient is reanalyzed to determine the effect of only four MSSVs per SG being available. The analysis assumes a 3% tolerance for all the available MSSVs. The MSSV on each SG with the lowest nominal setpoint was assumed unavailable, and the Unit 2 model is used because of its higher thermal rating. The results of the calculation show that the peak pressures in the SGs are lower than 1210 psia, or 110% of the 1085 psig SG design pressure.

Thus, with one MSSV inoperable per SG, the remaining MSSVs are capable of providing sufficient pressure relief capacity for the plant to operate at 100% RATED THERMAL POWER (RTP). However, the value applied to the high neutron flux trip setpoints must be lowered an additional 6% RTP to account for instrument and channel uncertainties (Ref. 7). This adjustment results in a setpoint of 94% RTP; however, the setpoint will remain at 87% RTP for additional conservatism.

More than One MSSV Inoperable

For more than one MSSV on each loop inoperable, the following Westinghouse algorithm contained in NSAL 94-001 is used:

$$\text{Hi}\,\phi = (100/Q)\frac{(w_s h_{fg} N)}{K}$$

where:

$\text{Hi}\,\phi$ = Safety Analysis PR high neutron flux setpoint, percent

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), MWt

(B)

INSERT B - Continued

K	=	Conversion factor, 947.82 (Btu/sec)/MWt
w_s	=	Minimum total steam flow rate capability of the operable MSSVs on any one SG at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs per SG is three, then w_s should be a summation of the capacity of the operable MSSVs at the highest operable MSSV operating pressure, excluding the three highest capacity MSSVs.
h_{fg}	=	heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm
N	=	Number of loops in plant

For the case of two and three inoperable MSSVs per SG, the setpoints derived are 53% and 35% RTP, respectively. However, the values applied to the high neutron flux trip setpoints must be lowered an additional 6% RTP to account for instrument and channel uncertainties (Ref. 7), which results in setpoints of 47% and 29% RTP, respectively.

(B)

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

The Power Range Neutron Flux-high trip setpoint must also be reduced in 72 hours (Required by Action A.2) to less than or equal to the value specified in Table 3.7.1-1, corresponding to the number of OPERABLE MSSVs regardless of the value of the MTC. Required Action A.2 is modified by a Note. The Note indicates that the Power Range Neutron Flux-high trip setpoint reduction is only required in MODE 1. In MODE 1, a reduced Power Range Neutron Flux-high trip setpoint provides the required protection. In MODES 2 and 3, the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation" provide sufficient protection. Thus, reduction of the Power Range Neutron Flux-high trip setpoint is not necessary in MODE 2 or 3.

The algorithm and the RETRAN analysis used for References 7 and 9 are conservative since they both assume that the relief capacity is accordingly reduced on each SG and that the flow capacity for all inoperable MSSVs is that of the highest capacity valve.

The calculated power level is further reduced to account for instrument and channel and heat balance uncertainties and is the value specified as the MAXIMUM ALLOWABLE % RTP in Table 3.7.1-1. Per Reference 7, the calculated instrument and channel uncertainties for the power range neutron flux measurement requires a further reduction of 6% RTP to assure that the maximum RTP is not exceeded with inoperable MSSVs. Therefore, when reducing the Power Range Neutron Flux-high trip setpoint, the setpoint must be reduced to less than or equal to the % RTP value shown on Table 3.7.1-1.

e

The allowed Completion Time~~[s]~~ are reasonable base on operating experience to complete the Required Action~~[s]~~ in an orderly manner without challenging unit systems.

~~For each steam generator, at a specified pressure, the fractional relief capacity (FRC) of each MSSV is determined as follows:~~

$$\cancel{FRC = \frac{A}{B}}$$

~~where:~~

~~A=the relief capacity of the MSSV; and~~

~~B=the total relief capacity of all the MSSVs of the steam generator.~~

~~The FRC is the relief capacity necessary to address operation with reduced THERMAL POWER.~~

~~The reduced THERMAL POWER levels in the LCO prevent operation at power levels greater than the relief capacity of the remaining MSSVs. The reduced THERMAL POWER is determined as follows:~~

(B)

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± ~~[3]~~% setpoint (as-found lift point) tolerance on the valves for OPERABILITY (with the exception of the lowest set MSSV setpoint which is (+3%/-2%); however, the valves are reset to ± 1% during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section ~~[~~10.3.1~~]~~.

2. Draft ASME Nuclear Pump and Valves Code Class II, 1968
~~2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.~~

3. FSAR, Section ~~[~~15.2~~]~~ and 15.3.

4. NRC Information Notice IN-94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

4 5. ASME, Boiler and Pressure Vessel Code, Section XI.

5 6. ANSI/ASME OM-1-1987 (including OM-a-1988 ADDENDA).

7. Westinghouse Report WCAP-11082, Revision 5, "Westinghouse Setpoint Methodology for Protection Systems Diablo Canyon Stations Eagle 21 Version", dated May 1993.
Units 1 and 2, 24 month Fuel Cycle Program

8. PG&E Design Calculation N-114, "Over-Pressure Study for One MSSV Per Loop Unavailable", dated 3/10/94.

9. PG&E Design Calculation N-115, "Reduced Power Levels for A Number of MSSVs Inoperable", dated 3/14/94.

(E)

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.

spell out main steam

One MSIV is located in each main steam line outside, but close to, containment. The MSIVs are installed back to back with the MS reverse flow check valves. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam isolation signal generated by ~~either~~ high negative steam line pressure rate or low steam line ~~generator~~ pressure or high-high containment pressure. The MSIVs are held in the open position and will fail in the closed direction on loss of control air ~~or~~ and fail open on loss of actuation power.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the FSAR, Section [10.3] (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the FSAR, Section ~~[6.2]~~ 6, Appendix 6.2 C (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the FSAR, Section ~~[15.1.5]~~ 15.4.2 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

The limiting case for the containment pressure analysis is the SLB inside containment, with initial reactor power at 30% with ~~a~~ no loss of offsite power, ~~following turbine trip,~~ and failure of the MSIV on the affected steam generator to close. At lower powers, the steam generator inventory and temperature are

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.4 10% Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The 10% ADVs (PCV-19, PCV-20, PCV-21 and PCV-22) provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Bypass System to the condenser not be available, as discussed in the FSAR, Section ~~[10.3]~~ 15 (Ref 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST) and firewater storage tank (FWST). The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Dump System.

One ADV line for each of the four steam generators is provided. Each ADV line consists of one ADV and an associated manual block valve.

The ADVs are provided with upstream manual block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

The ADVs are ~~usually~~ normally provided with a non-Class I pressurized ~~gas~~ supply of ~~bottled nitrogen~~ air ~~that, on~~ With a loss of pressure in the normal ~~instrument~~ air supply the backup non-Class I nitrogen supply, automatically supplies nitrogen to operate the ADVs. With the loss of both the normal air supply and the backup nitrogen supply, the normal supplies are blocked and the Class I backup air bottle system is activated. With the backup air bottle system activated, control of the valves is remote manual via the Class I control circuit from the Control Room. The ~~nitrogen~~ bottled air supply is sized to provide ~~the~~ sufficient pressurized gas to operate the ADVs for the time required for Reactor Coolant System cooldown to RHR entry conditions. In addition, handwheels are provided for local manual operation.

~~A description of the ADVs is found in Reference 1. The ADVs are OPERABLE with only a DC power source available. In addition, handwheels are provided for local manual operation.~~

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions at the maximum allowable~~. The design~~ rate of ~~[75]~~ 100°F per hour ~~is applicable for two steam generators, each with one ADV.~~ ~~This rate is adequate to cool the unit to RHR entry conditions with~~

(continued)

C

C.1

With three or more ADV lines inoperable, action must be taken to restore all but two ADV lines to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines.

~~C~~ D.1 and ~~C~~ D.2

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within [18] hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

Plant procedures which provide a 31 day verification that the 10% ADV manual block valves are open assures that the valves have not been inadvertently closed.

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened and closed ~~either~~ remotely ~~or locally and throttled through their full range~~ using the remote manual controls and the backup air bottles. This SR ensures that the ADVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ADV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components ~~usually~~ are expected to pass the Surveillance when performed at the ~~18~~ 24 month Frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.2

While not a safety function, the function of the manual block valve is to isolate a failed open ADV or isolate an ADV for repair or testing during plant operation. Cycling the block valve both closed and open demonstrates its capability to perform this function. ~~Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement.~~

(E)

BASES

b. Loss of MFW (the coincident loss of offsite power is a less limiting transient since RCP heat input is lost).

In addition, the minimum available AFW flow and system characteristics ~~are serious considerations~~ must be considered in the analysis of normal cooldown and ~~of a~~ small break loss of coolant accident (LOCA) due to their potential impact.

The AFW System is also designed for decay heat removal following a Steam Generator Tube Rupture (SGTR). As such the steam turbine driven AFW pump has redundant steam supplies to assure continued availability following a SGTR or MSLB event.

The AFW System design is such that it can perform its function following an FWLB between the MFW isolation valves and containment on loss of MFW, combined with a loss of offsite power following turbine trip, and a single active failure of the steam turbine driven AFW pump. ~~In such a case, the ESFAS logic may not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly.~~ One motor driven AFW pump would deliver to the broken MFW header at the pump ~~runout~~ maximum flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine driven pump ~~and associated power operated valves and controls~~ when required to ensure an adequate feedwater supply to the steam generators during loss of power. ~~DC~~ Vital AC power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary. ~~[~~Three~~]~~ independent AFW pumps in ~~[~~three~~]~~ diverse trains are required to be OPERABLE to ensure the availability of ~~RHR~~ decay and residual heat removal capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses~~. The~~ and having the third AFW pump ~~is~~ powered by ~~a different means.~~ a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs. To assure steam turbine driven AFW pump operability via redundant steam supplies, steam traps 104, 105 and 106 on the supply lines must be operable or bypassed to

(continued)

C

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The ~~18~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The ~~18~~ month Frequency is acceptable based on operating experience and the design reliability of the equipment.

This SR is modified by a Note that states the SR is not required in MODE 4 when the steam generator is being relied upon for heat removal. In MODE 4, the required AFW train ~~is~~ may already be aligned and operating.

24

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required. The ~~18~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR for the turbine-driven pump can be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. ~~[The]~~ Note 2 states that the SR is not required in MODE 4. In MODE 4, the required motor-driven pump is already operating and the autostart function is not required. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.

(continued)

(3)

BASES

and removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. ~~This~~ Decay heat removal may be during a normal or post accident cooldown and shutdown.

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is for one CCW ~~train~~ loop to remove the post ~~loss of coolant accident (LOCA)~~ DBA heat load from the containment ~~sump during the recirculation phase~~, without exceeding the design basis continuous ~~a maximum~~ CCW temperature of ~~[120]°F (Ref. 2). The Emergency Core Cooling System (ECCS) LOCA and containment OPERABILITY LOCA each model the maximum and minimum performance of the CCW System, respectively. The normal temperature of the CCW is [80]°F, and, during unit cooldown to MODE 5 (T_{cold} < [200]°F), a maximum temperature of 95°F is assumed. This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA, and provides a gradual reduction in the temperature of this fluid as it is supplied to the Reactor Coolant System (RCS) by the ECCS pumps.~~ ~~132°F and~~ not to exceed 120°F with an allowable transient not to exceed 140°F for more than 6 hours (Ref. 1).

(no direct correlation to 1967 GDC, however intent of 1971 GDC is met per FSAR Appendix 3.1A)

In accordance with GDC 44, the CCW system is designed to provide sufficient heat removal for normal and post accident ESF heat loads without overheating. The CCW system and ASW system are essentially considered a single heat removal system for the purpose of assessing the ability to sustain either a single active or passive failure and still perform design basis heat removal. Only one ASW pump and one CCW heat exchanger is required, as assumed in the safety analysis, to provide sufficient heat removal from containment to mitigate a DBA. However, to ensure maximum heat removal capability, operators are instructed to place the second CCW heat exchanger in service early in the emergency operating procedures.

The CCW System also functions to cool the unit from RHR entry conditions ($T_{coldave}$ < 350°F), to MODE 5 ($T_{coldave}$ < 200°F), during normal and post accident operations. The time required to cool from 350°F to 200°F is a function of the number of CCW heat exchangers and RHR ~~trains~~ heat exchangers operating. ~~One CCW train exchanger is sufficient to remove decay heat during subsequent operations with T_{cold} < [200]°F. This assumes a maximum service water temperature of [95]°F occurring simultaneously with the maximum heat loads on the system.~~

In the event that CCW system leakage occurs and system makeup is not available, the surge tank volume provides a minimum of 20 minutes, based on a non-mechanistic leakage rate of 200 gpm, for operators to locate and isolate the leak or realign the CCW

(continued)

E

BASES

system into two separate loops before the system becomes impaired due to water loss.

The CCW System satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

In the event of a DBA, one CCW loop is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water. To ensure this requirement is met, two vital loops of CCW must be OPERABLE. At least one CCW loop will operate assuming the worst case single active failure occurs coincident with a loss of offsite power. To meet the LCO on Component Cooling Water loops, vital headers A and B, both CCW heat exchangers, the surge tank, and all three CCW pumps must be operable.

A vital CCW loop is considered OPERABLE when:

a. Two CCW pumps, one CCW heat exchanger, and the surge tank are OPERABLE; and

b. The associated piping, valves, ~~heat exchanger~~, and instrumentation and controls required to perform the safety related function are OPERABLE.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System except for isolation of CCW to the CFCUs. Isolation of CCW to the CFCUs could potentially affect the flow balance and requires evaluation to ensure continued operability.

Split loop alignment of the CCW system during normal operation requires Condition A to be entered because the CCW system cannot tolerate a single failure in this configuration.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be prepared to perform its principal safety related function of removal of accident generated containment heat via the CFCUs and removal of decay heat from the reactor via the Residual Heat Removal (RHR) System.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

(continued)

C

proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The ~~18~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~18~~ 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated safety related actuation signal. ~~The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation.~~ The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This surveillance requirement applies to the SIS auto-start and the 4KV auto-transfer automatic starts only. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~18~~ 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 9.2.2.
2. FSAR, Section 6.2.

C

BASES

SR 3.7.8.3

This SR verifies proper automatic operation of the ~~SWS~~ ASW pumps on an actual or simulated safety related actuation signal. The ~~SWS~~ ASW is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The ~~[18]~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This surveillance requirement applies to the SIS auto-start and the 4kV auto transfer automatic starts only. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~[18]~~ 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section ~~[9.2.1]~~ 9.2.7.
2. FSAR, Section 6.2.
3. ~~FSAR, Section [5.4.7]~~ NRC Generic Letter 91-13, "Request for Information Related to the Resolution of Generic Issue 130, 'Essential Service Water System Failures at Multi-unit Sites,' Pursuant to 10 Cfr 50.54 (F)," dated September 19, 1991.

E

BASES

In MODE 5 or 6, the OPERABILITY requirements of the UHS are determined by the systems it supports.

ACTIONS

A.1

If ~~one or more cooling towers have one fan~~ the UHS is inoperable (i.e., ~~up to one fan per cooling tower inoperable~~ inlet water temperature > 64°F), the compensatory action ~~must be taken to restore the inoperable cooling tower fan(s) UHS to OPERABLE status within 7 days 8 hours by~~ of placing a second CCW heat exchanger in service must be performed within 8 hours. This action provides assurance that the ASW system and the CCW system can operate within its temperature limit.

The ~~7 day~~ 8 hour Completion Time is reasonable based on the low probability of an accident occurring during the ~~7 days~~ 8 hours that ~~one cooling tower fan is inoperable (in one or more cooling towers), the number of available systems,~~ the temperature is > 64°F without two CCW heat exchangers in service and the time required to reasonably complete the Required Action.

B.1 and B.2

If the ~~cooling tower fan~~ the second heat exchanger cannot be ~~restored to OPERABLE status~~ placed in service within the associated Completion Time, ~~or] if the UHS is inoperable for reasons other than Condition A,~~ the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.9.1 Not used

~~This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the SWS pumps. The [24] hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is [662] ft [mean sea level].~~

E

BASES

SR 3.7.9.2

~~This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the UHS is [90 F].~~

This SR verifies that adequate long term (30 day) cooling can be maintained. The 24, 12 and 2 hour surveillance Frequencies are based on operating experience related to trending of the temperature variations during the applicable MODES. This SR verifies the temperature of the UHS so that appropriate actions can be taken to assure that the ASW system can continue to assure that the CCW system will not exceed its design temperature profile.

SR 3.7.9.3 Not Used

~~Operating each cooling tower fan for ≥ [15] minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.~~

SR 3.7.9.4 Not Used

~~This SR verifies that each cooling tower fan starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with the typical refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.~~

REFERENCES

1. FSAR, Section ~~[~~9.2.5~~]~~.
2. ~~Regulatory Guide 1.27.~~ FSAR, Sections 2.4.11.5 & 2.4.11.6.
3. AEC Safety Guide 27.
4. ~~SSER 16.~~

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room ~~Emergency Filtration~~ Ventilation System (CRE~~F~~VS)

BASES

BACKGROUND

The CRE~~F~~VS provides a protected environment from which operators can control the units from the common control room following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The CRE~~F~~VS consists of two independent, redundant trains that recirculate and filter the control room air (one train from each unit). Each train consists of a heater, a prefilter ~~or demister~~, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and ~~a~~ one pressurization supply fan, one filter booster fan, and one main supply fan. Ductwork, ~~valves or~~ dampers, and instrumentation also form part of the system~~, as well as demisters to remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank~~.

The CRE~~F~~VS is an emergency system, parts of which may also operate during normal unit operations. ~~in the standby mode of operation.~~ Upon receipt of ~~the~~ an actuating signal~~(s)~~, the normal air supply to the control room is isolated, and the stream of outside ventilation air from the pressurization system and ~~is~~ recirculated control room air is passed through the system filter ~~trains~~. The pressurization system draws outside air from either the north end or the south end of the turbine building based upon the wind direction or the absence of releases at the inlet. The prefilters ~~or demisters~~ remove any large particles in the air, ~~and any entrained water droplets present~~, to prevent excessive loading of the HEPA filters and charcoal adsorbers. Continuous operation of each filter train for at least 10 hours per month, with the heaters on, reduces moisture buildup on the HEPA filters and adsorbers. ~~Both~~ Tthe ~~demister and~~ heater is ~~are~~ important to the effectiveness of the charcoal adsorbers.

Manual or automatic actuation of the CRE~~F~~VS places the system in one of three ~~either of two separate~~ states: 1) pressurization (MODE 4), 2) recirculation (MODE 3), or 3) smoke removal (MODE 2). ~~(emergency radiation state or toxic gas isolation state) of the emergency mode of operation, depending on the initiation signal~~. Actuation of the system to the recirculation mode ~~emergency radiation state of the emergency mode of operation,~~ closes the unfiltered outside air intake and unfiltered exhaust dampers, and aligns the system for recirculation of the control room air through the redundant trains of HEPA and the charcoal filters. The pressurization mode ~~emergency radiation state~~ also

MODE 4 is the only required MODE for the CRVS to be considered OPERABLE. The other MODES of operation are useful for certain emergency situations, such as control room smoke removal; but they are not required for CRVS operability.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Once actuated due to a fuel handling accident the CRVS must be protected against a single failure. This protection, although not required for immediate accident response, is assured by requiring that a backup power supply be provided as described above in Applicability. This back up is assured via the performance of non-TS surveillances that verify the ability to transfer power supplies.

The 31 day procedural verification of the separate vital power supplies for the redundant fans and the one hour operation of each supply, booster and pressurization supply fan (unless already operating) assures system reliability and two train redundancy.

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Monthly heater operations dry out any moisture accumulated in the charcoal from humidity in the ambient air. ~~[~~Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized and operating automatically (filter temperature control). ~~Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.]~~ The 31 day Frequency is based on the reliability of the equipment and the two train redundancy availability.

SR 3.7.10.2

This SR verifies that the required CR~~EF~~VS testing is performed in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. The CR~~EF~~VS filter tests are in accordance with ~~Regulatory Guide 1.52~~ ANSI 510-1980 (Ref. 3). The ~~[~~VFTP~~]~~ includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~.

SR 3.7.10.3

This SR verifies that each CR~~EF~~VS train automatically starts and operates in the pressurization mode on an actual or simulated actuation signal generated from a Phase "A" Isolation. The Frequency of ~~[18]~~ 24 months is specified in ~~Regulatory Guide 1.52~~ ANSI 510-1980 (Ref. 3).

(continued)

C

BASES

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CR~~EF~~VS. During the pressurization ~~emergency~~ mode of operation, the CR~~EF~~VS is designed to pressurize the control room ≥ ~~[0.125]~~ inches water gauge positive pressure with respect to the outside atmosphere and adjacent areas in order to prevent unfiltered inleakage. The CR~~EF~~VS is designed to maintain this positive pressure with one train ~~at a makeup flow rate of [3000] cfm~~. The Frequency of ~~[18]~~ 24 months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. 4).

REFERENCES

1. FSAR, Section 9.4.1 ~~[6.4]~~.
2. FSAR, Chapter 15.
3. ~~Regulatory Guide 1.52, Rev. 2~~ ANSI 510-1980.
4. NUREG-0800, Section 6.4, Rev. 2, July 1981.
5. DCM S-23F

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BASES

supplies for the exhaust fans, via a non-TS surveillance, assures system redundancy. ~~Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.]~~ The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

SR 3.7.12.2

This SR verifies that the required ~~ECCS PREACS~~ ABVS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The ~~ECCS PREACS~~ ABVS filter tests are in accordance with References 3 and 4. The VFTP includes testing HEPA filter performance, charcoal adsorber~~s~~ efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.12.3

This SR verifies that each ~~ECCS PREACS~~ ABVS train starts and operates on an actual or simulated actuation signal and that the system aligns to exhaust through the common HEPA filter and charcoal adsorber. The ~~18~~ 24 month Frequency is consistent with that specified in References 3 and 4.

SR 3.7.12.4 Not Used

~~This SR verifies the integrity of the ECCS pump room enclosure. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the ECCS PREACS. During the [post accident] mode of operation, the ECCS PREACS is designed to maintain a slight negative pressure in the ECCS pump room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The ECCS PREACS is designed to maintain a ≤ [-0.125] inches water gauge relative to atmospheric pressure at a flow rate of [3000] cfm from the ECCS pump room. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 6).~~

~~This test is conducted with the tests for filter penetration; thus, an [18] month Frequency on a STAGGERED TEST BASIS is consistent with that specified in Reference 4.~~

SR 3.7.12.5 Not Used

~~Operating the ECCS PREACS bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the ECCS PREACS bypass damper is verified if it can be specified in Reference 4.~~ /

BASES

SURVEILLANCE REQUIREMENT (continued)

SR 3.7.12.6

This SR verifies the leak tightness of dampers that isolate flow to the normally operating filter train. This SR assures that the flow from the auxiliary building passes through the HEPA filter and charcoal absorber unit when the ABVS Buildings and Safeguards or Safeguards Only modes have been actuated coincident with an SI. The ~~18~~ 24 month Frequency is consistent with the requirements of Reference 4.

REFERENCES

1. FSAR, Section ~~[6.5.1]~~ 9.4.2.
~~2. FSAR, Section [9.4.5].~~
~~3~~ 2. FSAR, Section ~~[15.6.]~~15.5.
~~4~~ 3. ~~Regulatory Guide 1.52 (Rev. 2).~~ ASTM D 3803-1989
4. ANSI N510-1980 .
5. 10 CFR 100.11.
6. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
7. DCM S-23B, "Main Auxiliary Building Heating and Ventilation System".

BASES

APPLICABLE SAFETY ANALYSES

The ~~FBACS~~ FHBVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference ~~3~~ 2, assumes that all fuel rods in an assembly are damaged. ~~The analysis of the LOCA assumes that radioactive materials leaked from the Emergency Core Cooling System (ECCS) are filtered and adsorbed by the FBACS.~~ The DBA analysis of the fuel handling accident assumes that only one train of the ~~FBACS~~ FHBVS is functional due to a single failure that disables the other train. The accident analysis accounts for the reduction in airborne radioactive material provided by the one remaining train of this filtration system. The amount of fission products available for release from the fuel handling building is determined for a fuel handling accident ~~and for a LOCA~~. In accordance with assumptions made in the fuel handling accident analysis, loss of offsite power is not considered concurrent with a fuel handling accident. However, loss of power is enveloped by the fuel handling accident analysis. To maximize FHBVS capability to mitigate the consequences of a fuel handling accident, at least one of the FHBVS trains must be capable of being supplied from an operable emergency diesel generator at all times whenever fuel movement is taking place in the spent fuel pool. These assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. ~~4~~ 3).

The ~~FBACS~~ FHBVS satisfies Criterion 3 of 10 CFR 50.36 (c) (2) (ii).

LCO

In accordance with assumptions made in the fuel handling accident analysis, loss of offsite power is not considered concurrent with a fuel handling accident. However, loss of power is enveloped by the fuel handling accident analysis.

Two independent and redundant trains of the ~~FBACS~~ FHBVS are required to be OPERABLE to ensure that at least one train is available, assuming a single failure that disables the other train~~, coincident with a loss of offsite power~~. This requires that when two trains of the FHBVS are OPERABLE, at least one train of the FHBVS must be capable of being powered from an OPERABLE diesel generator that is directly associated with the bus which energizes the FHBVS train. When only one train is OPERABLE, an OPERABLE diesel generator must be directly associated with the bus which energizes that one OPERABLE FHBVS train. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR 100 (Ref. ~~5~~ 4) limits in the event of a fuel handling accident.

The ~~FBACS~~ FHBVS is considered OPERABLE when the individual components necessary to control ~~exposure in the~~ releases from fuel handling building are OPERABLE in both trains. An ~~FBACS~~ FHBVS train is considered OPERABLE when its associated:

a. Exhaust ~~Fan~~ fan is OPERABLE;

(continued)

C

BASES

SR 3.7.13.3

This SR verifies that each ~~FBACS~~ FHBVS train starts and operates on an actual or simulated actuation signal and directs its exhaust flow through the HEPA Filters and charcoal absorber banks. The ~~18~~ 24 month Frequency is consistent with Reference 6.

SR 3.7.13.4

This SR verifies the integrity of the fuel handling building enclosure. The ability of the fuel handling building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested ~~to verify~~ proper function of the ~~FBACS~~ FHBVS. During the ~~[~~post accident~~]~~ mode of operation, the ~~FBACS~~ FHBVS is designed to maintain a slight negative pressure in the fuel handling building, to prevent unfiltered LEAKAGE. The ~~FBACS~~ FHBVS is designed to maintain ~~a~~ the building pressure ≤ -0.125 inches water gauge with respect to atmospheric pressure ~~at a flow rate of [20,000]cfm to the fuel building~~. The Frequency of ~~18~~ 24 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 7).

An 18 month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference 6.

SR 3.7.13.5 Not Used

~~Operating the FBACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FBACS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with Reference 6.~~

(continued)

E

BASES

This water level corresponds to 24 feet 6 inches above the top of the fuel assemblies in the racks and to 23 feet above a fuel assembly lying horizontally on top of the racks.

The spent fuel ~~storage~~ pool water level satisfies Criterion 2 of 10 CFR 50.36 (c) (2) (ii).

LCO

The spent fuel ~~storage~~ pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the fuel storage pool.

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the spent fuel ~~storage~~ pool, since the potential for a release of fission products exists.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the spent fuel ~~storage~~ pool water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel ~~storage~~ pool is immediately suspended ~~to a safe position~~. ~~This action effectively precludes the occurrence of a fuel handling accident.~~ This does not preclude movement of a fuel ~~assemblyies~~ assembly to a safe position.

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

This SR is done during the movement of irradiated fuel assemblies as stated in the Applicability. This SR verifies sufficient fuel

(continued)

E

B 3.7 PLANT SYSTEMS

B 3.7.16 Spent Fuel ~~Storage~~ Pool Boron Concentration

BASES

BACKGROUND

~~In the Maximum Density Rack (MDR) [(Refs. 1 and 2)] design.~~ The spent fuel ~~storage~~ pool is divided into two separate and distinct regions which, for the purpose of criticality considerations, are considered as separate pools. ~~[~~Region 1~~]~~, with ~~[336]~~ 290 storage positions, ~~is designed to accommodate~~ has been analyzed for the storage of new fuel ~~with a maximum enrichment of [4.65] wt% U-235, or spent fuel regardless of the discharge fuel burnup~~ assemblies which meet the requirements of LCO 3.7.17.1. ~~[~~Region 2~~]~~, with ~~[2670]~~ 1034 storage positions, ~~is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domain according to Figure [3.7.17-1 2], in the accompanying~~ has been analyzed for the storage of fuel assemblies which meet the requirements of LCO 3.7.17.2. ~~Fuel assemblies not meeting the criteria of Figure [3.7.17] shall be stored in accordance with paragraph 4.3.1.1 in Section 4.3, Fuel Storage.~~

The water in the spent fuel ~~storage~~ pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, ~~specify~~ require that the limiting k_{eff} of the fuel configuration is at or below the limit of 0.95 ~~be evaluated~~ in the absence of soluble boron. Hence, the ~~design~~ analysis of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. ~~3~~1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. ~~For example, the most severe accident scenario is associated with the movement of fuel from [Region 1 to Region 2], and accidental misloading of a fuel assembly in [Region 2]. This could potentially increase the to criticality of [Region 2]. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the MDR with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.17, "Spent Fuel Assembly Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.16.1.~~

BASES

~~fuel assemblies in the spent fuel storage pool. This LCO does not apply following the verification, since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movements in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly~~.

ACTIONS

A.1, ~~A.2.1,~~ and A.2~~.2~~

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the spent fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of fuel assemblies and immediately taking actions to restore the spent fuel pool boron concentration to greater than or equal to 2000 ppm. ~~The concentration of boron is restored simultaneously with suspending movement of fuel assemblies. An acceptable alternative is to verify by administrative means that the fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored.~~ This suspension of fuel movement does not preclude movement of ~~a~~ fuel assembl~~y~~ies to a safe position.

STET

If the LCO is not met while moving ~~irradiated~~ fuel assemblies ~~in MODE 5 or 6,~~ LCO 3.0.3 would not be applicable~~. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore,~~ since the inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.16.1

This SR verifies by chemical analysis that the concentration of boron in the spent fuel ~~storage~~ pool is ~~within~~ at or above the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The ~~7~~ 31 day Frequency is appropriate because no major replenishment of pool water is expected to take place ~~over such a short period of time~~.

(continued)

(E)

JUSTIFICATION FOR DIFFERENCES FROM NUREG 1431

NUREG-1431 Section 3.7

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

CHANGE NUMBER	JUSTIFICATION
3.7-01	ACTION A.1 is revised, new ACTION A.2 is added, & [and NSAL 94-001] and Table 3.7.1-1 is revised consistent with Traveler WOG-83, Rev. 0 to account for the fact that a reduction in power level is not directly proportional to the reduction in main steam safety valves (MSSV) relieving capability and plants which may operate for some part of a fuel cycle with a positive moderator temperature coefficient (MTC). Per Westinghouse Nuclear Safety Advisory Letter, NSAL 94-001, if the MTC is positive at the required reduced power level, the reactor coolant system (RCS) heat up following a turbine trip event could result in a core power increase and additional heat transfer to the secondary system which may not be attenuated without over pressurizing the main steam system. To preclude this condition the power range neutron flux high trip set point is required to be reset to a power level consistent with the number of inoperable safety valves within 72 hours. [A Note is ~~added that states~~ that Required Action A.2 is only applicable in MODE 1]. These changes are consistent with Westinghouse Owners Group (WOG) Traveler WOG-83 and NSAL 94-001. And Table 3.7.1-1 is revised to delete reference to five OPERABLE MSSVs A recent revision to WOG-83 (Rev.1) has been proposed requiring that the power range neutron flux trip high setpoints be reduced when at a reduced reactor power level to account for a control rod withdrawal event at reduced reactor power. The identification of this issue has identified a non-conservatism in NUREG-1431. Consequently, the requirement in the CTS to reduce the power range neutron flux trip high setpoints with inoperable MSSVs [regardless of the value of MTC] is retained. However, the 72 hour Completion Time proposed in the traveler is incorporated into the ITS. These changes are acceptable because the retention of the requirement to reduce the power range flux trip high setpoint is more conservative than NUREG-1431 or WOG-83 and the extended Completion Time recognizes the low probability of an event occurring during the 72 hours allowed to reset the trip setpoints.
3.7-02	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B)
3.7-03	SR 3.7.3.1 is divided into two surveillances since both the stroke time and the surveillance Frequency requirements are different for the feedwater regulation and associated bypass valves and the feedwater isolation valves.
3.7-04	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B)

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CHANGE NUMBER	JUSTIFICATION
3.7-26	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-27	A Note is added to Table 3.7.1-2 under LIFT SETTING that specifies that the lift point of the lowest set safety is +3% and -2%.
3.7-28	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-29	Revise AFW pump testing Frequency to be "In accordance with Inservice Test Program." These changes are consistent with TSTF-101, and will eliminate any ambiguity associated with pump testing as a result of ASME changes.
3.7-30	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-31	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B)
3.7-32	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B)
3.7-33	The requirement to verify a make-up flow rate during the tests demonstrating the capability to maintain control room differential pressure above atmospheric pressure would be deleted. The current licensing basis of the plant is to be able to maintain a positive pressure in the control room with respect to the outside atmosphere.
3.7-34	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-35	SR 3.7.10.3 is revised to reflect plant configuration and current licensing basis required testing.
3.7-36	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B)
3.7-37	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-38	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-39	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-40	Not used.
3.7-41	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-42	This change adds a Note that states that LCO 3.0.3 is not applicable to the fuel handling building ventilation system during fuel movement since fuel movement is independent of reactor operation. This exemption is part of the DCPP CTS and has been proposed as a generic change to NUREG-1431 by Industry Traveler TSTF-36, Rev. 2.
3.7-43	ACTION A is revised and ACTIONS ~~C~~ B and ~~E~~ C are not used, and ACTION D is relabeled as B per the current licensing basis. The FHBVS for the plant does not act as part of the ventilation system used to filter post LOCA leakage external to the containment.

(E)

CHANGE NUMBER	JUSTIFICATION
3.7-44	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-45	Not applicable to DCPP. See Conversion Comparison Table (Eenclosure 6B).
3.7-46	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-47	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-48	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).
3.7-49	The requirement to verify a make-up flow rate during the tests demonstrating the capability to maintain [fuel handling] building differential pressure below atmospheric pressure would be deleted. The current licensing basis of the plant is to be able to maintain a negative pressure [in the fuel handling building] with respect to the outside atmosphere.
3.7-50	The CTS ADV surveillance that verifies the back-up air bottle pressure once per 24 hours is retained.
3.7-51	A new spent fuel pool storage specification is created for Region 1 fuel storage due to the unique storage requirements.
3.7-52	ITS 3.7.11 is not used due to the mild coastal environment in which DCPP is located, consistent with the CTS.
3.7-53	ITS 3.7.16 is revised to be consistent with the current licensing basis and CTS. The boron concentration is required to be within limits whenever fuel is stored in the spent fuel pool to prevent an increase in the k_{eff} of the racks above 0.95 should the spent fuel pool temperature increase above 150°F. The Frequency for verification of the boron concentration is changed from 7 days to 31 days consistent with the CTS.
3.7-54	The LCO, Required Actions, and Surveillances are revised per the CTS. The CTS evaluates Region 2 fuel storage on fuel pellet diameter and a checker board loading pattern in addition to the other NUREG-1431 requirements.
3.7-55	The NUREG-1431 3.7.1~~4~~9 specification is not used since an equivalent safety grade system does not exist. Therefore, the deletion is per the current licensing basis.
3.7-56	This change creates a new SR for the MSIVs [and MFIVs , MFRVs and associated bypass valves] to distinguish between the IST and the automatic actuation testing of these isolation valves. The surveillance allows credit for an actual actuation, if one occurs, to satisfy the surveillance requirements. These changes are consistent with WOG-98. Although SRs 3.7.2.2 and 3.7.3.2 are new SRs, they may be performed in conjunction with SRs 3.7.2.1 and 3.7.3.1. Therefore, the Note allowing testing to be performed ~~on~~ in MODE 3 is also needed for ~~these new~~ SR~~s~~ 3.7.2.1
3.7-57	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 6B).

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-37	Modifies LCO 3.7.2 Condition A and adds new Condition B and C to be consistent with the CPSES CTS.	No	Yes	No	No
3.7-38	This proposed change deletes reference to a specific flowrate for conducting the negative pressure test per the CPSES CTS.	No, see CN 3.7-49.	Yes	No, see CN 3.7-49.	No, see CN 3.7-49.
3.7-39	SR 3.7.12.6 is added to verify the shutdown of the non-ESF fans to prevent bypass of the ESF Filtration units (CPSES specific).	No	Yes	No	No
3.7-40	Not used.	N/A	N/A	N/A	N/A
3.7-41	The Main Feedwater Regulating and associated Bypass Valves are deleted from the ITS per current licensing basis.	No, CTS includes MFRVs.	No, refer to 3.7-11.	Yes	Yes
3.7-42	Add DCPP specific note that states that 3.0.3 is not applicable to the fuel handling building ventilation system during fuel movement since fuel movement is independent of reactor operation.	Yes	No	No	No
3.7-43	ACTION A of ITS 3.7.13 is revised and ACTIONS ~~C,~~ E B and F. C of ITS 3.7.13 are not used and ACTION D is Relabled as B per the DCPP CTS.	Yes	No	No	No
3.7-44	This change would revise ITS 3.7.13 to add a new Note to the Applicability and change the Conditions, Required Actions, and SRs to conform to the design of the Emergency Exhaust System.	No, fuel building ventilation not required for post LOCA leakage.	No, CTS does not require this specification.	Yes	Yes

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-45	ITS 3.7.15 is revised to be CPSES specific to address the two spent fuel pools and the in-containment storage racks.	No	Yes	No	No
3.7-46	Revised to delete "irradiated fuel assemblies seated in" since accident analysis assumes fuel assembly lying on top of the fuel storage racks.	No, ITS is consistent with CTS.	Yes	No, ITS is consistent with CTS.	Yes
3.7-47	This change adds TS 3.7.19, a safety chilled water system which is in the CPSES CTS.	No	Yes	No	No
3.7-48	This change adds TS 3.7.20, an UPS HVAC system which is in the CPSES CTS.	No	Yes	No	No
3.7-49	The requirement to verify a make-up flow rate during the tests demonstrating the capability to maintain [fuel handling] building differential pressure below atmospheric pressure would be deleted. The current licensing basis of the plant is to be able to maintain a negative pressure [in the fuel handling building] with respect to the outside atmosphere.	Yes	No, see CN 3.7-38.	Yes	Yes
3.7-50	The CTS DCPP specific ADV surveillance that verifies the back-up air bottle pressure once per 24 hours is retained.	Yes	No	No	No
3.7-51	A new spent fuel pool storage specification is created for Region 1 fuel storage due to unique storage requirements at DCPP.	Yes	No	No	No
3.7-52	ITS 3.7.14 is not used due to the mild coastal environment in which the plant is located.	Yes	No	No	No
3.7-53	ITS 3.7.16 for DCPP is revised to be consistent with the current licensing basis and CTS.	Yes	No	No	No

TECHNICAL SPECIFICATION CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
3.7-54	The LCO, Required Actions, and Surveillances are revised per the DCPP specific CTS to incorporate Region 2 fuel storage requirements.	Yes, per LA 116/114.	No	No	No
3.7-55	NUREG-1431 Specification 3.7.14 is not used since an equivalent safety grade system does not exist. Therefore, the deletion is per the current licensing basis.	Yes	No	No	No
3.7-56	This change creates a new SR for the MSIVs [and MFIVs] to distinguish between the IST and the automatic actuation testing of these isolation valves. The SR allows credit for an actual actuation, if one occurs, to satisfy the surveillance requirements. Although SRs 3.7.2.2 and 3.7.3.2 are new SRs, they may be performed in conjunction with SRs 3.7.2.1 and 3.7.3.1. Therefore, the Note allowing testing to be performed ~~on~~ in MODE 3 is also needed for these new SRs. 3.7.2.1.	Yes	Yes	Yes	Yes
3.7-57	This change establishes appropriate Required Actions and Completion Times for ventilation system pressure envelope degradation.	{No, retained CTS.}	Yes	Yes	Yes

(11)

DCL-97-106, LAR 97-09
ITS 3.8 ERRATA

SR 4.8.3.1.f

(E)

LS14
02-21-~~LS1~~
02-05-M
02-18-LG
02-04-M

f. At least once per 18 months during shutdown, by giving performance discharge tests, or a modified performance discharge test, of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application.~~ Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating. ~~; and at least once per 24 months when battrery has reached 85% of service life for the application with capacity ≥100% of manufacture's rating; this surveillance shall not be performed in Mode 1, 2, 3, or 4~~, but credit may be taken for unplanned events that satisfy this SR~~.

~~*The resistance of cell to cell connecting cables does not have to be included.~~

02-24-LG

B

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.14 -------------------NOTES------------------ 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not be performed in MODE 1 or 2. ~~However, credit may be taken for unplanned events that satisfy this SR.~~ -- Verify each DG operating at a power factor ≤ 0.87 ~~[0.9]~~ operates for ≥ 24 hours: a. For ≥ 2 hours loaded ≥ 2625 ~~[6250]~~ kW and ≤ 2890 ~~[5500]~~ kW; and 2860 b. For the remaining hours of the test loaded ≥ 2370 ~~[4500]~~ kW and ≤ 2610 ~~[5000]~~ kW.	3.8-20 B 18 months B-PS B B-PS B-PS
SR 3.8.1.15 -------------------NOTES------------------ 1. This Surveillance shall be performed within 5 minutes of shutting down the DG after the DG has operated ≥ 2 hours loaded ≥ 2370 ~~[4500]~~ kW and ≤ 2610 ~~[5000]~~ kW. Momentary transients outside of load range do not invalidate this test. 2. All DG starts may be preceded by an engine prelube period. -- Verify each DG starts and achieves a. in ≤ 10 seconds, speed ≥ 900 rpm; and b. in ≤ 13 ~~[10]~~ seconds, voltage ≥ 3785 ~~[3740]~~ V, and ≤ 4400 ~~[4580]~~ V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	B B-PS 18 months 3.8-40 B B-PS B

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, ~~and~~ Starting Air, and Turbocharger Air Assist

BASES

BACKGROUND

~~Each diesel generator (DG) is provided with a storage tank having a fuel oil capacity sufficient to operate that diesel for a period of 7 days while the DG is supplying maximum post loss of coolant accident load demand discussed in the FSAR, Section [9.5.4.2] (Ref. 1). The maximum load demand is calculated using the assumption that a minimum of any two DGs is available. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time to replenish the onsite supply from outside sources.~~

~~Fuel oil is transferred from storage tank to day tank by either of two transfer pumps associated with each storage tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve or tank to result in the loss of more than one DG. All outside tanks, pumps, and piping are located underground.~~

50,000

The diesel fuel oil storage system consists of two common tanks with a nominal capacity of ~~40,000~~ gallons each. The TS-required fuel oil quantity is based on the calculated fuel oil consumption necessary to support the operation of the DGs to power the minimum engineered safety feature (ESF) systems required to mitigate a design basis accident (LOCA) in one unit and those minimum required systems for a concurrent non-LOCA safe shutdown in the remaining unit (both units initially in Mode 1 operation). The fuel oil consumption is calculated for a period of 7 days operation of minimum ESF systems. This requirement provides a sufficient operating period within which offsite power can be restored and/or additional fuel can be delivered to the site.

Fuel oil is transferred from the storage tanks via the diesel fuel oil storage and transfer

BASES

system to replenish the day tanks as required. The design incorporates sufficient redundancy so that a malfunction of either an active or a passive component will not impair the ability of the system to supply fuel oil. Two redundant fuel oil transfer pumps supply fuel oil to DG day tanks from either storage tank. One pump is adequate to supply the six DGs operating at full load. Each DG tank has two separate, redundant transfer pump start-stop level switches. Each level switch automatically starts a transfer pump and opens the supply header solenoid valve corresponding to the respective transfer pump, 0-1 or 0-2. In addition, high and low level alarms are provided on each day tank and activate alarms both locally and in the control room.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the fuel oil. Regulatory Guide 1.137 (Ref. 2) addresses the recommended fuel oil practices as supplemented by ANSI N195 (Ref. 3). The fuel oil properties governed by these SRs are the water and sediment content, the kinematic viscosity, specific gravity (or API gravity), and impurity level.

INSERT A

The DG lubrication system is designed to provide sufficient lubrication to permit proper operation of its associated DG under all loading conditions. The system is required to circulate the lube oil to the diesel engine working surfaces and to remove excess heat generated by friction during operation. ~~Each engine oil sump contains an inventory capable of supporting a minimum of [7] days of operation. [The onsite storage in addition to the engine oil sump is sufficient to ensure 7 days of continuous operation.]~~ The diesel lube oil storage requirement is based upon a conservative usage factor of 1% of fuel oil consumption. The storage system used to meet this requirement is that located in the warehouse where 650 gallons of lube oil is stored in drums. This storage is augmented by a second storage location within the diesel engine itself. The lube oil level on each engine's dip stick is maintained 5 inches above the engine's operability limit. This provides approximately 120 gallons of usable lube oil within each of the 6 diesel engines. INSERT B This supply is sufficient to allow the operators to replenish lube oil from outside sources, As a third resource

INSERT A: The DG fuel oil consumption is calculated for a period of 7 days operation of minimum ESF systems. This requirement provides a sufficient operating period within which offsite power can be restored and/or additional fuel can be delivered to the site.

INSERT B: The total engine oil sump inventory (all engines) is capable of supporting a minimum of 7 days of operation at minimum ESF loads. The onsite storage inventory (warehouse) is in addition to the engine oil sump also sufficient to ensure 7 days of continuous operation.

A

BASES

APPLICABLE SAFETY ANALYSES

Each DG has ~~an~~ two redundant 100% capacity air start systems and a turbocharger air assist system with adequate capacity for ~~five~~ three successive start attempts each on the DG without recharging the air start receiver~~(s)~~ ~~or the turbocharger air assist air receiver.~~ TP

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter ~~[6]~~ (Ref. 4), and in the FSAR, Chapter ~~[15]~~ (Ref. 5), assume Engineered Safety Feature (ESF) systems are OPERABLE. The DGs are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that fuel, Reactor Coolant System and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power Distribution Limits; Section 3.4, Reactor Coolant System (RCS); and Section 3.6, Containment Systems.

Since diesel fuel oil, lube oil, ~~and the~~ air start, and turbocharger air assist subsystems support the operation of the standby AC power sources, they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) the NRC Policy Statement.

LCO

Stored diesel fuel oil is required to have sufficient supply for 7 days of ~~full load~~ minimum ESF systems operation. The required combined stored diesel fuel oil is a contained quantity with different storage requirements for unit operation in MODE 1, 2, 3, and 4 and for MODE 5 and 6. With both units operating in MODE 1, 2, 3, and 4, the required level is ≥ 65,000 gallons. With one unit operating in MODE 1, 2, 3, or 4, and the other unit in MODE 5 or 6, the required fuel oil level is 33,000 gallons plus 26,000 gallons, for a total of 59,000 gallons combined storage. With both units in MODE 5 or 6, the required fuel oil level is 52,000 gallons. The required combined stored fuel oil

(Continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.4

THREE

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for a minimum of ~~[five]~~ ~~six~~ engine start cycles without recharging. ~~[A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed.]~~ Each start cycle is 15 seconds of cranking. The pressure specified in this SR is intended to reflect the lowest value at which ~~the [five]~~ three starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every ~~[~~31~~]~~ days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, ~~and~~ contaminated fuel oil, ~~and~~ or from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water

(Continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.6

three

This Surveillance ensures that, without the aid of the refill compressor, sufficient turbocharger air assist air receiver capacity for each DG is available. The system design requirements provide for a minimum of six engine start cycles without recharging. Each start cycle is 15 seconds of cranking. The pressure specified in this SR is intended to reflect the lowest value at which three starts can be accomplished.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal turbocharger air assist air receiver pressure.

~~Draining of the fuel oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning are required at 10 year intervals by Regulatory Guide 1.137 (Ref. 2), paragraph 2.f. This SR also requires the performance of the ASME Code, Section XI (Ref. 8), examinations of the tanks. To preclude the introduction of surfactants in the fuel oil system, the cleaning should be accomplished using sodium hypochlorite solutions, or their equivalent, rather than soap or detergents. This SR is for preventive maintenance. The presence of sediment does not necessarily represent a failure of this SR, provided that accumulated sediment is removed during performance of the Surveillance~~

REFERENCES

1. FSAR, Section 9.5.4.2.
2. Regulatory Guide 1.137.
3. ANSI N195-1976, Appendix B.
4. FSAR, Chapter 6.

(Continued)

Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

LCO 3.8.9 CONDITION A
4160 VAC and 480 VAC

VOLTAGE	BUS F MAJOR ESF LOADS (TRAIN A)	BUS G MAJOR ESF LOADS (TRAIN B)	BUS H MAJOR ESF LOADS (TRAIN A&B)
4160 VAC	ASW PP 1 AFW PP 3 CCP PP 1 CCW PP 1 SI PP 1 480 VAC BUS F	ASW PP 2 CS PP 1 RHR PP 1 CC PP 2 CCW PP 2 480 VAC BUS G	AFW PP 2 (B) CS PP 2 (A) RHR PP 2 (A) SI PP 2 (B) CCW PP 3 (A&B) 480 ~~VAV~~ VAC BUS H
480 VAC *	CFCU 1 CFCU 2	CFCU 3 CFCU 5	CFCU 4 (A&B)

* Partial listing of loads

LCO 3.8.9 CONDITION B
120 VAC

BUS 1 PY11 (21)** PY11A (21A)**	BUS 2 PY12 (22)**	BUS 3 PY13 (23)** PY13A (23A)**	BUS 4 PY14 (24)**
IY Powered by: 480 VAC BUS F/DC BUS 1 or TRY1 Powered by: 480 VAC BUS F or Backup 480 VAC BUS G	IY1 Powered by: 480 VAC BUS G/DC BUS 2 or TRY2 Powered by: 480 VAC BUS G or Backup 480 VAC BUS F	IY Powered by: 480 VAC BUS H/DC BUS 3 or TRY3 Powered by: 480 VAC BUS H or Backup 480 VAC BUS G	IY Powered by: 480 VAC BUS H/DC BUS 2 or TRY1 Powered by: 480 VAC BUS H or Backup 480 VAC BUS F

** Unit 2 in parentheses

DCL-97-106, LAR 97-09
ITS 3.9 ERRATA

3/4.9.12 FUEL HANDLING BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two Fuel Handling Building Ventilation ~~Systems~~ trains shall be OPERABLE.

APPLICABILITY: ~~Whenever~~ During movement of irradiated fuel ~~is~~ assemblies in the ~~spent~~ Fuel ~~pool~~ Handling Building. 12-01-LS24

ACTION:

a. With one Fuel Handling Building Ventilation ~~System~~ train inoperable, fuel movement within the spent fuel pool ~~or crane operation with loads over the spent fuel pool~~ may proceed provided the OPERABLE Fuel Handling Building Ventilation ~~System~~ train is ~~capable of being powered from an OPERABLE emergency power source and is~~ in operation ~~and discharging through at least one train of HEPA filters and charcoal absorbers~~. 12-02-LG 12-09-LG 12-04-A

b. With no Fuel Handling Building Ventilation ~~System~~ trains OPERABLE, suspend all operations involving movement of irradiated fuel within the spent fuel pool ~~or crane operation with loads over the spent fuel pool~~ until at least one Fuel Handling Building Ventilation ~~System~~ train is restored to OPERABLE status. 12-02-LG ED

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

~~4.9.12 The above required Fuel Handling Building Ventilation Systems shall be demonstrated OPERABLE:~~

a. At least once per 31 days by initiating flow through each train of FHBVS ~~the HEPA filters and charcoal absorbers~~ and verifying that the system operates for at least 15 minutes; ED 12-04-A

b. At least once per ~~18 months~~ REFUELING INTERVAL or (1) after any structural maintenance on the HEPA filter or charcoal absorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Visually verifying that, with the system operating at a flow rate of 35,750 cfm ± 10% and exhausting through the HEPA filters and charcoal absorbers, the damper valve M-29 is closed;

   2) Verifying that the cleanup system satisfies the in place penetration and bypass Leakage testing acceptance criteria of less than 1% and uses the test procedures guidance in ANSI N 510 - 1980, and the system flow rate is 35,750 cfm ± 10%; 12-04-A

C

3) ~~Verifying a system flow rate of 35,750 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980.~~ 12-11-A

c. At least once per 18 months or (1) after any structural maintenance on the charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system, or (3) after every 720 hours of charcoal adsorber operation by verifying ~~within 31 days after removal,~~ that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 at 95% R.H. for a methyl iodide penetration of less than 4.3%; 12-10-LS9

d. At least once per REFUELING INTERVAL by: 12-06-A

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4.1 inches Water Gauge while operating the system at a flow rate of 35,750 cfm ± 10%,

2) Verify~~ing~~ that on ~~a high radiation test~~ actual or simulated actuation signal, ~~the system automatically starts (unless already operating),~~ and directs its exhaust flow through the HEPA filters and charcoal adsorber banks, and 12-05-TR1 12-04-A

3) On a STAGGERED TEST BASIS, Verify~~ing~~ that ~~the system~~ a train maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during system operation. each item of the ~~train~~ train activates 12-08-LS16

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a DOP test aerosol while operating the system at a flow rate of 35,750 cfm ± 10%; and 12-04-A

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon test gas while operating the system at a flow rate of 35,750 cfm ± 10%.

C

DESCRIPTION OF CHANGES TO TS SECTION 3/4.9
(Continued)

CHANGE NUMBER	NSHC	DESCRIPTION
11-04	LG	This change moves the restriction on crane operation to a licensee controlled document. The restriction on crane operations may be removed because it is not in the assumptions used for the FHA. Crane operations that could adversely affect fuel stored in the spent fuel pool are controlled as analyzed in the review of heavy load movements. This change is consistent with NUREG-1431, and moves requirements that do not meet the criteria for inclusion in the TS.
12-01	LS24	The Applicability would be changed to "During movement of irradiated fuel in the fuel building" instead of "Whenever irradiated fuel is in the spent fuel pool" consistent with NUREG-1431. The proposed Applicability is consistent with the assumptions used in the FHA in the Fuel Handling Building which postulates the inadvertent drop of an irradiated fuel assembly. Potential damage to fuel assemblies due to dropping of heavy loads is addressed by CN 12-02-LG.
12-02	LG	Moves the restriction on crane operations over the spent fuel storage areas when the fuel building air cleanup system was inoperable. The restriction on crane operations may be removed because it is not consistent with the assumptions used for the FHA. Crane operations that could adversely affect fuel stored in the spent fuel pool is prohibited in accordance with plant procedures as analyzed in the review of heavy load movements.
12-03	A	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-04	A	The SR regarding filter testing would be moved to a "Ventilation Filter Testing Program" that is called out in the Administrative Controls Section 5.5.11 of the ITS. This change does not result in a change to technical requirements.
12-05	TR1	Revised SR to allow for increased flexibility in using an actual or simulated actuation signal. Identification of the specific signal is moved to the Bases.
12-06	A	This requirement would have the operability of each train of the [Fuel Handling Building Ventilation System (FHBVS)] (including maintaining negative pressure in the building) to be demonstrated. This is consistent with current practice. This change does not result in a change to technical requirements and is consistent with NUREG-1431.
12-07	LS25	Not applicable to DCPP. See Conversion Comparison Table (Enclosure 3B).
12-08	LS16	The proposed change would allow the ~~18~~ [24]-month testing of the [FHBVS] ability to maintain the required pressure differential between the building and the outside atmosphere to be performed on a STAGGERED TEST BASIS.

TECH SPEC CHANGE		APPLICABILITY			
NUMBER	DESCRIPTION	DIABLO CANYON	COMANCHE PEAK	WOLF CREEK	CALLAWAY
12-08 LS16	The proposed change would allow the ~~18~~ [24]-month testing of the [FHBVs] ability to maintain the required pressure differential between the building and the outside atmosphere to be performed on a STAGGERED TEST BASIS.	Yes	No, CPSES does not have this specification	Yes	Yes
12-09 LG	The requirement for an OPERABLE emergency power source for an OPERABLE FHBV train is moved to the Bases.	Yes	No, CPSES does not have this specification.	No	No
12-10 LS9	The "within 31 days after removal" requirement for completion of laboratory analyses is deleted.	Yes	No, CPSES does not have this specification in CTS 3/4.9.	Yes	Yes
12-11 A	The SR to measure [FHBVs] flow rate is not identified as a separate SR in the ITS because it is verified during the other in-place filter tests (see ITS 5.5.11 a. and b.)	Yes	No, CPSES does not have this specification in CTS 3/4.9.	Yes	Yes
12-12 LS26	This change establishes appropriate ACTIONS and Completion Times for Fuel Building pressure envelope degradation.	{No, maintaining CTS}	No, CPSES does not have this specification in CTS 3/4.9	Yes	No, maintaining CTS
14-01 LS11	This change deletes the restrictions on placing spent fuel assemblies into Region 2 of the spent fuel pool and changing storage locations designations from Region 1 to Region 2.	No, Requirement not in CTS.	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes
14-02 M	This changes the Applicability from "Whenever irradiated fuel assemblies are in the spent fuel pool" to "Whenever any fuel assembly is in Region 2 of the spent fuel pool."	No, already in CTS.	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes
14-03 LS12	This change would delete the ACTION requirements to suspend all other movement of spent fuel and crane operations.	Yes	No, CPSES does not have this specification in CTS 3/4.9	Yes	Yes

C

IV. SPECIFIC NO SIGNIFICANT HAZARDS CONSIDERATIONS

NSHC LS16
10CFR 50.92 EVALUATION FOR
TECHNICAL CHANGES THAT IMPOSE LESS RESTRICTIVE
REQUIREMENTS WITHIN THE TECHNICAL SPECIFICATIONS

The proposed change would allow the ~~18~~ [24]-month testing of the FHBVS ability to maintain the required pressure differential between the building and the outside atmosphere to be performed on a STAGGERED TEST BASIS in accordance with NUREG-1431. The current requirement Frequency is simply ~~18~~ [24] months. The new STS define STAGGERED TEST BASIS such that one train of the system would be tested every ~~36~~ [48] months rather than the ~~18~~ [24] months required in the current Specification. This is a relaxation in testing requirements.

This proposed TS change has been evaluated and it has been determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92© as quoted below:

> *"The Commission may make a final determination, pursuant to the procedures in 50.91, that a proposed amendment to an operating license for a facility licensed under 50.21(b) or 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:*
>
> 1. *Involve a significant increase in the probability or consequences of an accident previously evaluated; or*
>
> 2. *Create the possibility of a new or different kind of accident from any accident previously evaluated; or*
>
> 3. *Involve a significant reduction in a margin of safety."*

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

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

   The FHBVS is an accident mitigation system; it is not a precursor to any previously analyzed accident. Therefore, the proposed change to the frequency of testing would not affect the probability of an accident occurring. The proposed change would reduce the frequency of testing the building pressure reduction feature of the system. However, the system would still be tested for operability by (1) monthly operation, (2) ~~18~~ [24] month automatic start, and (3) testing under the Ventilation Filter Test Program. Also, the factors that affect building pressurization, in addition to the capability of the fan, do not change significantly over time and intentional changes to them are performed under administrative controls, These factors include alternations to the building pressure envelope (piping and electrical penetrations, and doors). Since these are controlled (a retest of the exhaust system would be required if they changed significantly) and since the fan operability is checked monthly, the proposed change would still provide assurance that the FHBVS remains OPERABLE. Also, primary degradation of the exhaust fans normally occurs over an extended period. Operating experience based on maintaining similar equipment further supports the acceptability of the proposed test interval. Thus, the proposed reduced frequency of testing does not significantly increase the consequences of an accident. Therefore, the proposed change would have an insignificant effect on the consequences of a previously evaluated accident.

E

BASES (continued)

ACTIONS (continued)

~~The Completion Time of 4 hours is sufficient to obtain and analyze~~ a ~~reactor coolant sample for boron concentration.~~ The Completion Time ~~Frequency~~ of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions. For core reload, the first CANNEL CHECK for each channel may be performed using the first fuel assembly as a source, prior to unlatching it in the core. X

CHANNEL

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The CHANNEL CALIBRATION also includes verification of the audible alarm and count rate functions on a simulated or actual boron dilution flux doubling signal. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.

2. FSAR, Section ~~15.4.6 [~~ 15.2.4 ~~]~~.

3. License Amendment 46/45, October, 1989.

B

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation—High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS), as required by GDC 34 (DCPP meets the intent of this 1971 GDC), to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg(s). Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass lines. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSIS

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to ~~a reduction in boron concentration in the coolant due to~~ boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission product barrier.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement , it was identified in ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii) as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat

(Continued)

(B)

BASES

core prior to 57 hours of core subcriticality. The second part of this Surveillance serves the same function but with 57 hours or more of core subcriticality. The flow rate of 1300 gpm is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. Both of these flow rates are points of the same flow rate verses decay heat. 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 (Ref. 2).

REFERENCES

1. FSAR, Section ~~[~~ 5.5.7 ~~]~~.
2. LAR 88-01, dated ~~4/21/88~~ 1/22/88 submitted by "RHR System Flow Rate Reduction," DCL ~~88-087~~ 88-014.

(B)

BASES

path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs. One or both RHR pumps maybe aligned to the RWST to support filling the refueling cavity or for performance of required testing (Ref. 2).

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level." A Note is added to the applicability to assure that MODE 6 operation with water level <23 ft is not permitted unless two RHR loops are operable.

ACTIONS

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

B.1

If no RHR loop is in operation, 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 the unborated water sources are isolated.~~ THE SUSPENSION OF ANY OPERATION INVOLVING A REDUCTION IN REACTOR COOLANT BORON CONCENTRATION WILL REDUCE THE LIKELIHOOD OF BORON STRATIFICATION IN THE RCS.

B.2

If no RHR loop is in operation, actions shall be initiated immediately, and continued, to restore one RHR loop to operation. Since the unit is in Conditions A and B concurrently, the restoration of two OPERABLE RHR loops and one operating RHR loop should be accomplished expeditiously.

B.3

If no RHR loop is in operation, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop

(Continued)

B

BASES

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 that 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.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that one RHR loop is in operation and circulating reactor coolant. The flow rate of more than 3000 gpm is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core prior to 57 hours subcritical. The second part of this Surveillance serves the same function but with 57 hours or more of core subcriticality and provides a reduced flow rate of 1300 gpm based upon a reduced decay heat load. Both of these flow rates are points of the same flow rate verses decay heat curve. The 1300 gpm limit also precludes exceeding the 1675 gpm upper flow limit to prevent vortexing and air entrainment of the RHR piping system. RHR pump vortexing (failure to meet pump suction requirements) during mid-loop operation may result in RHR pump failure and non-conservative RCS level indication. ~~In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR pump suction requirements must be met.~~ The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator for monitoring the RHR System in the control room (Ref. 3).

SR 3.9.6.2

Verification that the required pump is OPERABLE ensures that an additional ~~RCS or~~ RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

1. FSAR, Section ~~[~~ 5.5.7 ~~].~~ 1/22/88
2. WOG Standard Technical Specification Change Traveler TSTF-21
3. LAR 88-01, dated ~~4/21/88~~, submitted by "RHR System Flow Rate Reduction," DCL ~~88-067~~ 88-014

(Continued)

DCL-97-106, LAR 97-09
ITS 4.0 ERRATA

~~DESIGN PRESSURE AND TEMPERATURE~~ 02-01-LG

~~5.2.2 Containment is designed and shall be maintained for a maximum internal pressure of 47 psig and a temperature of 271°F, coincident with a Double Design Earthquake.~~

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy-4 or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with ~~NRC-~~approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analysis to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core locations. 03-01-A

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 ~~full length and no part length~~ control rod assemblies. The ~~full length~~ control rod ~~assemblies~~ material shall ~~contain a nominal 142 inches of absorber material. The nominal values of absorber material shall~~ be ~~80%~~ silver, ~~15%~~ indium, ~~and 5%~~ and cadmium as approved by the NRC. ~~All control rods shall be clad with stainless steel tubing.~~ 03-02-~~A~~ LG

~~5.4 REACTOR COOLANT SYSTEM~~

~~DESIGN PRESSURE AND TEMPERATURE~~

~~5.4.1 The Reactor Coolant System is designed and shall be maintained:~~ 04-01-LG

- ~~a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements.~~
- ~~b. For a pressure of 2485 psig, and~~
- ~~c. For a temperature of 650°F, except for the pressurizer which is 680°F.~~

~~VOLUME~~

~~5.4.2 The total water and steam volume of the Reactor Coolant System is 12,811 ± 100 cubic feet at a nominal T_{avg} of 576°F for Unit 1 and 12,903 ± 100 cubic feet at a nominal T_{avg} of 577°F for Unit 2.~~ 04-01-LG

(E)

DESCRIPTION OF CHANGES TO TS SECTION 5.0

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

CHANGE NUMBER	NSHC	DESCRIPTION
01-01	LG	Figures representing site location/exclusion area boundary and low population zone replaced by text description to be consistent with NUREG-1431.
01-02	LG	Map for radioactive gaseous and liquid effluents moved to a licensee controlled document consistent with NUREG-1431.
02-01	LG	Containment design description moved to licensee controlled document consistent with NUREG-1431.
03-01	A	The description of the fuel assemblies was reworded to be consistent with NUREG-1431. Proposed rewording does not involve any technical changes.
03-02	~~A~~ LG	Detailed information regarding control rod construction moved to a licensee controlled document. A reworded general description of the control rods is provided consistent with NUREG-1431.
03-03	A	Not Applicable to DCPP. See Conversion Comparison Table. (Enclosure 3B)
04-01	LG	The description of the reactor coolant system volume and pressure and temperature limits is removed from the Technical Specifications (TS). This information is consistent with information already contained in the licensee controlled documents. The change is consistent with NUREG-1431.
05-01	LG	The meteorological tower location is removed from the TSs. This information is consistent with information already contained in licensee controlled documents. The change is consistent with NUREG-1431.
06-01	A	The fuel storage - criticality section is reformatted consistent with NUREG-1431. The proposed reformatting does not involve any technical changes.
06-02	M	The new fuel storage section is revised consistent with NUREG-1431. Proposed revisions provide details regarding analysis assumptions/limitations for the storage of new fuel (equivalent to that provided for spent fuel storage). The additional details are also consistent with licensee controlled documents.
06-03	A	Not Applicable to DCPP. See Conversion Comparison Table. (Enclosure 3B)