Pacific Gas and Electric Company

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March 16, 2000

PG&E Letter DCL-00-041

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
License Amendment Request 00-01,
Administrative Revisions to the Improved Technical Specifications

Dear Commissioners and Staff:

Enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 pursuant to 10 CFR 50.90. This license amendment request (LAR) proposes to change the Improved Technical Specifications (ITS), as approved under License Amendments 135 and 135 dated May 28, 1999, to correct errors made when incorporating final mark-ups to NUREG-1431 into the ITS. The changes are administrative in nature.

A description of the proposed ITS changes, and the bases for the changes, are provided in Enclosure A. The proposed ITS changes are noted on the marked-up copy in Enclosure B. The proposed ITS pages are provided in Enclosure 3.

The changes proposed in this LAR are not required to address an immediate safety concern. However, to assure correct implementation of the ITS, PG&E requests that the NRC review and approve this LAR to support the ITS implementation date of May 31, 2000.

Sincerely,

David H. Oatley

cc: Edgar Bailey, DHS
Steven D. Bloom
Ellis W. Merschoff
David Proulx
Diablo Distribution

Enclosures

A001

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY	)	Docket No. 50-275 Facility Operating License No. DPR-80
Diablo Canyon Power Plant Units 1 and 2	)	Docket No. 50-323 Facility Operating License No. DPR-82

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath says that he is Vice President - Diablo Canyon Operations and Plant Manager of Pacific Gas and Electric Company; that he is familiar with the content thereof; that he has executed LAR 00-01 on behalf of said company with full power and authority to do so; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

David H. Oatley
Vice President
Diablo Canyon Operations and Plant Manager

Subscribed and sworn to before me this 16th day of March, 2000.

Notary Public
State of California
County of San Luis Obispo

CHUCK MACKEY
Commission # 1204640
Notary Public - California
San Luis Obispo County
My Comm. Expires Dec 12, 2002

ADMINISTRATIVE REVISIONS TO THE IMPROVED TECHNICAL SPECIFICATIONS

A. DESCRIPTION OF AMENDMENT REQUEST

This License Amendment Request (LAR) revises the Improved Technical Specifications (ITS) approved under License Amendment (LA) 135/135 dated May 28, 1999 as follows.

1. Typographical corrections listed below are corrections based upon errors in the certified copy of the ITS that was submitted by DCL 99-072, dated May 19,1999 and revised by DCL 99-078, dated May 27, 1999.

   a. Technical Specification (TS) 1.2, Example 1.2-2, Required Action A.2 - Correct the misalignment of the logical connector.

   b. TS 2.1, Figure 2.1.1-1 - Correct spelling error and text misalignment.

   c. TS 3.1.3, Figure 3.1.3-1 - Revise the legend to state MTC-PCM/°F rather then MTC-PCM.

   d. Limiting Condition of Operation (LCO) 3.2.2, Required Action A 1.2.2. - Correct completion time misalignment.

   e. LCO 3.3.1, Condition C - Correct spelling error.

   f. LCO 3.3.1, Required Action E, Note - Note reads "Function 26" but should read "Function 2.b."

   g. LCO 3.3.1, Table 3.3.1-1 - Correct table number in heading.

   h. LCO 3.3.1, Table 3.3.1-1 Function 6 and 7 - Correct referenced page numbers.

   i. TS 3.3.1, Table 3.3.1-1, Function 22 - The required surveillance tests is corrected by deleting SR 3.3.1.17 and replacing it with SR 3.3.1.5 (as provided by response to request for additional information item (RAI) number DC 3.3-006 in DCL 99-063).

   j. LCO 3.3.2, Required Action O - Correct character from numerical "0" to a capital "O."

   k. LCO 3.3.2, Required Action P, Note - Note will read; "One additional channel may be bypassed for up to 4 hours for surveillance testing" (as provided in response to RAI Q 3.3-66 in DCL 98-167).

l. TS 3.4.6, SR 3.4.6.3, Frequency - The frequency will be revised to 7 days (as shown in NUREG-1431 and the current Technical Specifications (CTS)).

m. LCO 3.4.8, Note 1 - Text should include the phrase "per 8 hour period" (as shown in the final markup of NUREG-1431 in DOC 01-06-M).

n. LCO 3.4.14, Required Action, A.2.1 and A.2.2 - Correct the misalignment of logical connector.

o. TS 3.7.10, Surveillance Requirement (SR) 3.7.10.2 - Correct numbering of surveillance.

p. TS 3.7.13, SR 3.7.13.4 - The word "on" should be revised to "one."

q. TS 5.5.9.d.1.f.1) - Insert the word "being" between "are" and "applied."

r. TS 5.5.11.a, b, e - The phrase "operating cycle" will be revised to "24 months" consistent with the CTS Errata RAI, DC-ALL-001 (DCL 98-107, dated August 5, 1998, and DCL 98-144, dated October 16, 1998). These errata corrections incorporated LA 119/117 and LA 118/116 (24 Month Fuel Cycle) and were made to the CTS (4.7.5.1.c & d, 4.7.6.1.b, c, & d, 4.9.12.b & d) but were inadvertently overlooked in the relocated material in ITS 5.5.11. The relocated CTS material corrected by DC-ALL-001 uses the phrase "Refueling interval" which is defined in CTS Table 1-1 as 24 Months.

s. TS 5.7.2.d.1; Add word "dose" between "radiation" and "rates."

2. Editorial Change listed below reflects a change to a title applicable at the time LAs 135/135 was approved.

a. TS 5.7.2.a.1 - The "Shift Supervisor" title is changed to "Shift Manager."

3. Corrections listed below reflect changes to the conversion application and it's associated supplements.

a. TS 3.3.1, Table 3.3.1-1, Function 5 - SR 3.3.1.16 (Response Time Testing) will be added to both Mode 2 and Mode 3, 4, 5 (with Control Rods capable of withdrawal).

b. TS 3.3.1, Table 3.3.1-1, Function 16 - SR 3.3.1.15 will be removed from Function 16.a and SR 3.3.1.10 will be removed from Function 16.b.

c. TS 3.3.2, Table 3.3.2-1, Function 3.b.(3); SR 3.3.2.10 will be removed from Function 3.b.(3).

d. TS 3.3.3, SR 3.3.3.2 - Restore the note excluding neutron detectors from Channel Calibration.

e. TS 3.3.5, SR 3.3.5.2 and SR 3.3.5.3 - The frequency will be restored to the CTS value of 18 months.

f. LCO 3.4.13.d - The 1 gpm value provided in LCO 3.4.13.d will be deleted from the LCO and the remaining items renumbered.

g. TS 3.4.14, SR 3.4.14.1 - The exception list under Frequency will be expanded to include valves 8701 and 8702.

h. TS 3.7.1, Table 3.7.1-1 - The CTS 3.7.1, Table 3.7-1 note which states "Unless the Reactor Trip System breakers are in the open position." will be added to Table 3.7.1-1.

i. TS 5.1.2 - The phrase "(other than the Shift Technical Advisor)" will be deleted.

j. TS 5.5.9.d.2 - The phrase "and all tubes containing through wall cracks" will be struck.

k. TS 5.7.1.e - Add the phrase "or personnel continuously escorted by such individuals" after "protection procedures."

B. BACKGROUND

On June 2, 1997, PG&E submitted LAR 97-09, "Technical Specification Conversion License Amendment Request," which proposed an improved set of TS for Diablo Canyon Power Plant (DCPP), Units 1 and 2. Between June 1997 and May 1999, PG&E and the NRC resolved comments and questions from the NRC on the proposed ITS. On May 28, 1999, the NRC issued the ITS in LA 135/135, for Units 1 and 2, respectively, to be implemented no later than May 31, 2000.

As part of the development of the LAR and the ITS, numerous reviews of the markups and the ITS were performed. However, due to the size of the document and the number of changes, it was recognized that some errors would occur. As such, PG&E planned on this LAR to resolve any errors that were identified during preparation for implementation following issuance of the ITS. The

proposed LAR will provide assurance that the ITS, when implemented in May 2000, is accurate and correct.

The largest percentage of the errors identified occurred in ITS Section 3.3. This was the last section reviewed by the NRC, and was completed just prior to the final submittal of the ITS. Consequently, the least number of extra reviews were performed on this section. To assure no other errors are in Section 3.3, additional Engineering and Regulatory Services reviews were performed. These reviews identified no additional items.

C. JUSTIFICATION

The proposed changes will provide clarification of the ITS and assure their correct interpretation and implementation.

The proposed change to the Shift Supervisor title is an administrative change that will eliminate confusion between DCPP procedures and the ITS.

D. SAFETY EVALUATION

The proposed corrections are administrative in nature. They will provide no changes to the intent of LA 135/135 or the intended operation of the plant. The changes will clarify the specifications and assure that misinterpretations will not occur during plant operation.

1. Typographical Corrections

The proposed changes are administrative changes. They have no impact on the requirements of the TS. The proposed changes make the final version of the ITS consistent with the mark-ups of NUREG-1431 provided to the NRC from June 2, 1997, to May 27, 1999, and approved as part LA 135/135 dated May 28, 1999.

These revisions are acceptable because they bring the ITS into conformance with the final NUREG-1431 mark-ups submitted to the NRC. These errors occurred during the incorporation of comments and creation of the final clean copy of the ITS.

2. Editorial Revision

The proposed change is an administrative change. It will have no impact on the requirements of the TS. The proposed change is consistent with the final version of the ITS and with the mark-ups of NUREG-1431 provided to the NRC from June 2, 1997, to May 27, 1999, and approved as part LAs 135/135 dated May 28, 1999.

This change is acceptable because there is no reduction in responsibility or qualifications for the new title of "Shift Manager." The work functions and responsibilities of the individual previously identified as "Shift Supervisor" remain unchanged.

3. Corrections

Several instances of incorrect incorporation of the CTS into the ITS have been identified. The basis for the acceptability of each change is identified below:

a. TS 3.3.1, Table 3.3.1-1, Function 5 - SR 3.3.1.16 (Response Time Testing)

The restoration of response time testing is acceptable because it is consistent with the CTS and the approved safety analysis for plants without upper head injection (such as DCPP) as described in WCAP-9500, "Reference Core Report - 17X17 Optimized Fuel Assembly," May 1982.

The deletion of the response time testing in the ITS was inappropriate. The CTS require response time testing because the source range neutron flux (high setpoint) and power range neutron flux (high and low setpoints) are credited for mitigation of an inadvertent rod withdrawal accident in Mode 3. As such, it is appropriate that the requirement be in the TS.

b. TS 3.3.1, Table 3.3.1-1, Function 16

SR 3.3.1.15 will be removed from Function 16.a, "Low Auto-Stop Oil Pressure," and SR 3.3.1.10 will be removed from Function 16.b, "Turbine Stop Valve Closure." This would result in Function 16.a requiring a Channel Calibration and Function 16.b requiring a trip actuation device operational test (TADOT) (without verification of setpoint). As discussed in ITS Section 3.3, No Significant Hazards Consideration LS-53, the auto-stop oil pressure switches are subject to drift and a channel calibration would assure proper setpoint as well as function. The valve position switches on the turbine stop valves are not subject to drift since they are fixed in location when installed. They have no instrumentation features but rather only provide actuation. A channel calibration is not applicable to this type of equipment. A TADOT without requiring verification of setpoint will properly verify this actuation function. This was the original intent of the conversion.

The CTS requires only a TADOT (without verification of setpoint) for both Function 16.a and Function 16.b. The markups for the ITS

inadvertently retained both the channel calibration and the TADOT (without verification of setpoint) for each function. As discussed above, these tests are essentially redundant in intent yet both would be required every 24 months during a refueling outage. This change will eliminate this redundancy as well as inappropriate test elements. This change is acceptable since it would assure that each function was appropriately verified.

c. TS 3.3.2, Table 3.3.2-1, Function 3.b.(3)

The response time test, SR 3.3.2.10, will be removed from Function 3.b.(3). This test is not required in CTS Table 3.3-5, Function 7.b, Phase "B" Isolation. There are no change numbers which document this change. This is an inadvertent change. This time response is not credited. The Phase "A" containment high pressure is the credited response time for containment isolation (See CTS Table 3.3-5, Function 2.a.3).

d. TS 3.3.3, SR 3.3.3.2

The Note stating "Neutron detectors are excluded from CHANNEL CALIBRATION" included in NUREG-1431 for SR 3.3.3.2 will be restored. Implementation reviews showed that the exemption provided in NUREG-1431 for the neutron detectors is required to perform the channel calibration.

The neutron wide range detectors were added to the CTS by DOC 08-11-M to make the CTS consistent with NUREG-1431. However, the note for SR 3.3.3.2 that excluded these detectors from calibration was incorrectly deleted via JFD 3.3-109 per RAI Q 8-11 in DCL 98-167.

Since these neutron detectors are fission chamber type detectors (different from the normal NIS detectors), it was believed at the time of submittal that the calibration requirements of the PAMS neutron wide range detectors would be less restrictive. However, like all other neutron detectors only a functional check can be made on the detector since a calibration would require a set of high intensity check sources. Therefore, the note will be restored in accordance with NUREG-1431.

e. TS 3.3.5, SR 3.3.5.2 and SR 3.3.5.3

The frequency will be restored to the CTS value of 18 months. Justification for a 24 month frequency was never provided to the NRC.

f. LCO 3.4.13.d

This item will be deleted from the LCO. The 1 gpm limit for all steam generator (SG) leakage was removed per LAs 124/122, dated March 12, 1998. The 1 gpm value for all SGs provided in LCO 3.4.13.d was replaced by the more restrictive requirement of no more than "150 gallon per day primary to secondary LEAKAGE through any one stream generator" requirement of LCO 3.4.13.e. LCO 3.4.13.e should be renumbered as LCO 3.4.13.d.

g. TS 3.4.14, SR 3.4.14.1

The exception list identifying those valves to which the SR does not apply will be expanded to include valves 8701 and 8702. These motor-operated valves are residual heat removal suction valves and are equipped with control room position indication, inadvertent opening interlocks, and system high pressure alarms. These valves were inadvertently overlooked during the conversion mark-ups of CTS Table 3.4-1, Note *.

h. TS 3.7.1, Table 3.7.1-1

The CTS 3.7.1, Table 3.7-1, contains a note which states "Unless the Reactor Trip System breakers are in the open position." This note was removed by DOC 01-04-LS3 as part of TSTF-235. RAI Q 3.7.1-4 questioned the status of this TSTF and as part of PG&E's response this change was withdrawn and CTS adopted for the conversion. This withdrawal restored this note to the table but the RAI marked-up pages of the ITS inadvertently omitted restoration of this note. This note is a necessary part of the CTS and will be added to ITS Table 3.7.1-1 at this time.

i. TS 5.1.2

The phrase "(other than the Shift Technical Advisor)" will be removed since LAs 135/135, DOC 01-15-A removed the term, "Shift Technical Advisor," from the TS. This occurrence of the term was inadvertently left in the TS. Removal of the term is administrative and does not change the engineering expertise on shift requirement which is still intended in the ITS.

j. TS 5.5.9.d.2

The phrase "and all tubes containing through wall cracks" will be removed. This phrase was deleted by LA 124/122, dated March 12, 1998. This phrase was shown removed by RAI DC-3.4-003 in the

CTS markup but inadvertently overlooked in the ITS markup and final copy of the ITS.

k. TS 5.7.1.e

The phrase "or personnel continuously escorted by such individuals" after "protection procedures" was added to TS 5.7.2.e, but was inadvertantly omitted from TS 5.7.1.e (see RAI Q 5.2-1, Difference 5.7-2, DCL 98-134). This phrase is applicable to both CTS 6.12.1 (ITS 5.7.1.e) and CTS 6.12.2 (ITS 5.7.2.e).

These corrections are acceptable since they assure that the TS will be consistent with the accident analysis and the change justifications provided with the conversion to the ITS.

E. NO SIGNIFICANT HAZARDS EVALUATION

PG&E has evaluated the no significant hazards considerations (NSHC) involved with the proposed amendment, focusing on the three standards set forth in 10 CFR 50.92(c) as set forth 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 for a testing facility involves no significant hazards considerations, 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 changes are administrative in nature. They correct errors made while incorporating the current Technical Specifications (CTS) into the improved Technical Specifications (ITS), or errors made while creating the final copy of the ITS from the NRC reviewed mark-up of NUREG-1431. The

proposed change of the Shift Supervisor title to Shift Manager is administrative since it does not decrease the responsibilities of the individual.

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The proposed changes are administrative.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed changes are administrative in nature. They correct errors made while incorporating the CTS into the ITS, or errors made while creating the final copy of the ITS from the NRC reviewed mark-up of NUREG-1431. The proposed change of the Shift Supervisor title to Shift Manager is administrative since it does not decrease the responsibilities of the individual.

   There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The changes are administrative in nature so there are no new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   The proposed changes are administrative in nature. They correct errors made while incorporating the CTS into the ITS, or errors made while creating the final copy of the ITS from the NRC reviewed mark-up of NUREG-1431. The proposed change of the Shift Supervisor title to Shift Manager is administrative since it does not decrease the responsibilities of the individual.

   The proposed changes do not affect the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

   Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above safety evaluation, PG&E concludes that the changes proposed by this LAR satisfy the NSHC standards of 10 CFR 50.92(c), and accordingly a no significant hazards finding is justified.

G. ENVIRONMENTAL EVALUATION

PG&E has evaluated the proposed change and determined the change does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

MARKED-UP IMPROVED TECHNICAL SPECIFICATIONS

Remove Page	Insert Page
Page 1.2-2	Page 1.2-2
Page 2.0-2	Page 2.0-2
Page 3.1-6	Page 3.1-6
Page 3.2-5	Page 3.2-5
Page 3.3-1	Page 3.3-1
Page 3.3-3	Page 3.3-3
Page 3.3-12	Page 3.3-12
Page 3.3-13	Page 3.3-13
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Page 3.7-19	Page 3.7-19
Page 3.7-24	Page 3.7-24
Page 5.0-1	Page 5.0-1
Page 5.0-12	Page 5.0-12
Page 5.0-16	Page 5.0-16
Page 5.0-20	Page 5.0-20
Page 5.0-21	Page 5.0-21
Page 5.0-32	Page 5.0-32

Enclosure 3
PG&E Letter DCL-00-041

REVISED TECHNICAL SPECIFICATIONS PAGES

1.2 Logical Connectors

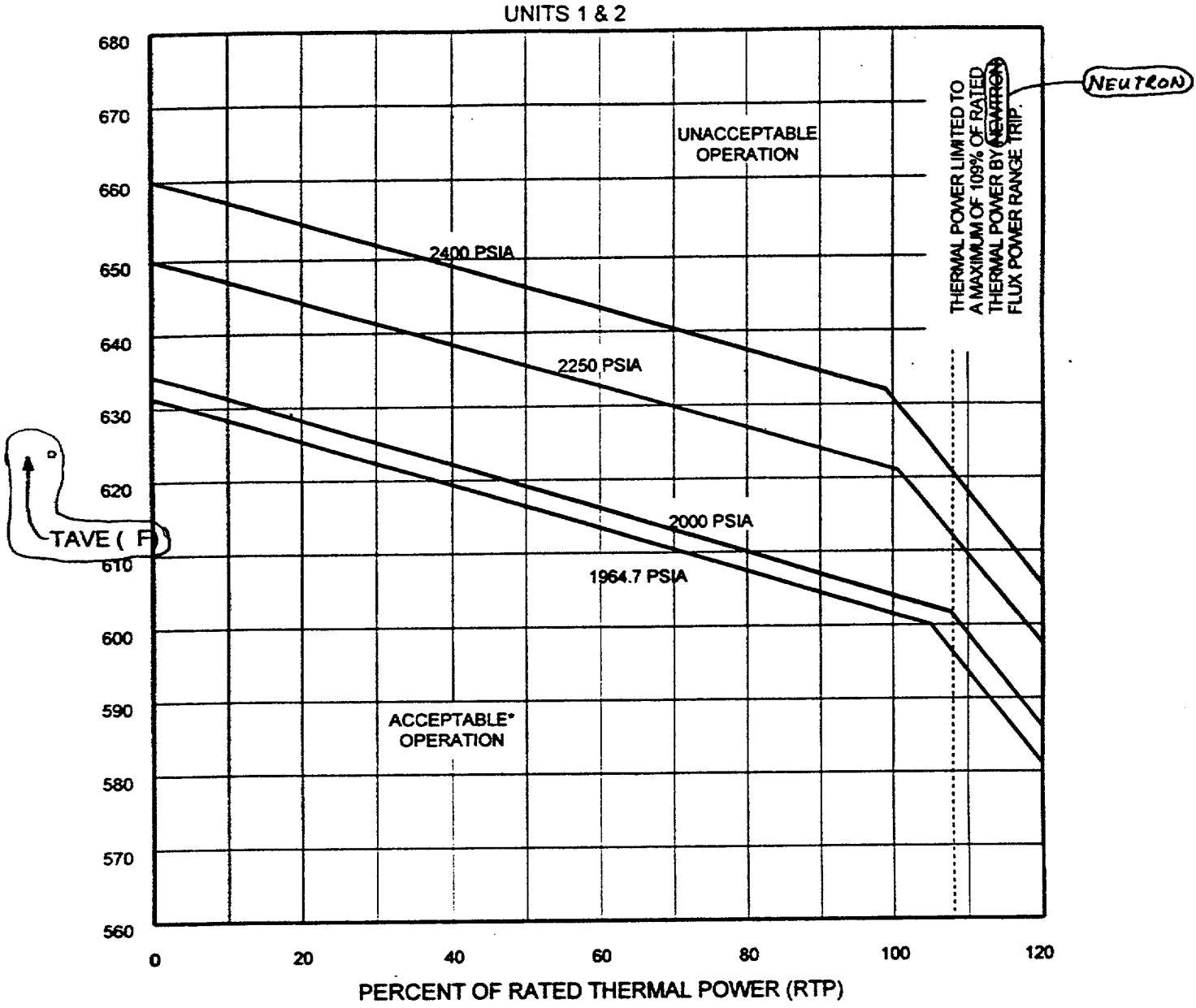
EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

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

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



*WHEN OPERATING IN THE REDUCED RTP REGION OF TECHNICAL SPECIFICATION LCO 3.4.1 (TABLE 3.4.1-1 FOR UNIT 1 AND TABLE 3.4.1-2 FOR UNIT 2) THE RESTRICTED POWER LEVEL MUST BE CONSIDERED 100% RTP FOR THIS FIGURE.

Figure 2.1.1-1

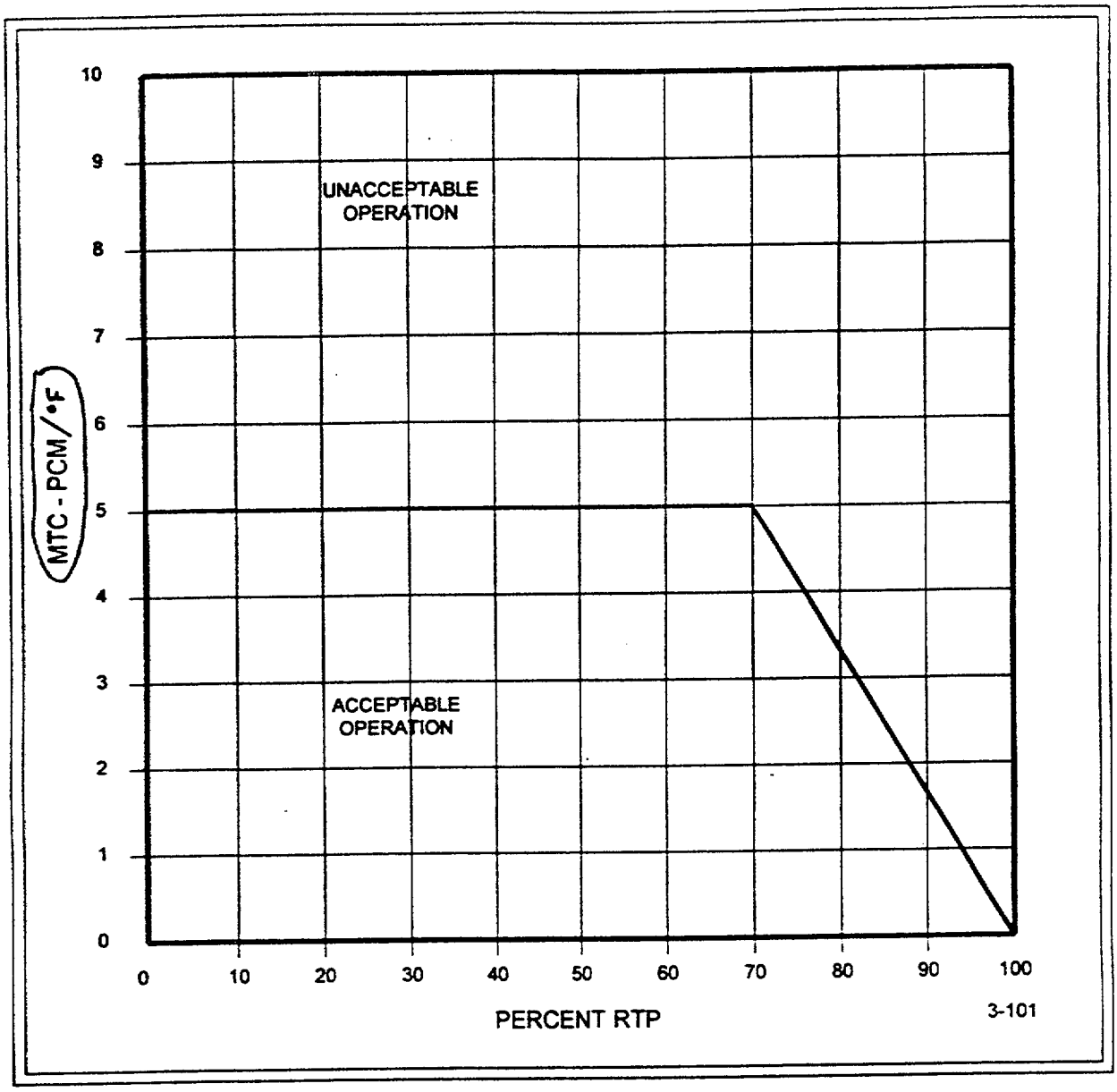


Figure 3.1.3-1 (page 1 of 1)

MODERATOR TEMPERATURE COEFFICENT vs. POWER LEVEL

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ————NOTE———— Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ——— $F^N_{\Delta H}$ not within limit.	A.1.1 Restore $F^N_{\Delta H}$ within limit.	4 hours
	OR	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND	72 hours
	A.1.2.2 Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP.	
	AND	
	A.2 Perform SR 3.2.2.1.	24 hours
	AND	(continued)

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

NOTE

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s) or trains.	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status. OR B.2 Be in MODE 3.	48 hours 54 hours
NOTE While this LCO is not met for function 19, 20 or 21, in MODE 5, making the Rod Control System capable of rod ~~withdraw~~ withdrawal is not permitted. C. One channel or train inoperable.	C.1 Restore channel or train to OPERABLE status. OR C.2.1 Initiate action to fully insert all rods.	48 hours 48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	----------NOTE---------- The inoperable channel, or one additional channel for functions 6, 7, and 8.b may be bypassed for up to 4 hours for surveillance testing of other channels. For functions 3.a, 3.b, 14.a ~~and 26~~ only the inoperable channel may be bypassed for surveillance testing of other channels. ----------	
	E.1 Place channel in trip. OR	6 hours
	E.2 Be in MODE 3.	12 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6. OR	24 hours
	F.2 Increase THERMAL POWER to > P-10.	24 hours
G. Two Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions. AND	Immediately
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
H. Not used		
I. One Source Range Neutron Flux channel inoperable.	I.1 Suspend operations involving positive reactivity additions.	Immediately
J. Two Source Range Neutron Flux channels inoperable.	J.1 Open reactor trip breakers (RTBs).	Immediately

(continued)

3.3.1-1

Table ~~3.3.3-1~~ (page 1 of 7)

Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA	NA
	3[b], 4[b], 5[b]	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 110.2% RTP	109% RTP
b. Low	1[c],2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 26.2% RTP	25% RTP
3. Power Range Neutron Flux Rate						
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 5.6% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 5.6% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1[c], 2[d]	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.6% RTP	25% RTP

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlocks.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

Table 3.3.1-1 (page 2 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
5. Source Range Neutron Flux	2[e]	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 1.4 E5 cps	1.0 E5 cps
	3[b], 4[b], 5[b]	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 1.4 E5 cps	1.0 E5 cps
	3[f], 4[f], 5[f]	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	N/A
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1 (Page 3.3-16)	Refer to Note 1 (Page 3.3-16)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 2 (Page 3.3-17)	Refer to Note 2 (Page 3.3-17)
8. Pressurizer Pressure						3.3-17 / 3.3-18
a. Low	1[g]	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1947.5 psig	1950 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2387.5 psig	2385 psig
9. Pressurizer Water Level—High	1[g]	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 90.2%	90%

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(f) With the RTBs open or all rods fully inserted and incapable of withdrawal. In this condition, source range Function does not provide reactor trip but does provide indication.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

3.3.1-1

Table ~~3.3.3-1~~ (page 3 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
10. Reactor Coolant Flow—Low	1[g]	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 89.8%[l] of MMF/ loop	90%[l] of MMF/loop
11. Reactor Coolant Pump (RCP) Breaker Position	1[g]	1 per RCP	M	SR 3.3.1.14	NA	NA
12. Undervoltage RCPs	1[g]	2 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 7877 V each bus	8050 V each bus
13. Underfrequency RCPs	1[g]	3 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 53.9 Hz each bus	54.0 Hz each bus
14. a. Steam Generator (SG) Water Level—Low Low	1,2	3 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 7.0%	7.2%
b. SG Water Level - Low Low Trip Time Delay (TTD)	1,2	4	X	SR 3.3.1.7 SR 3.3.1.10	TTD ≤ 1.01 TD (Note 3) for RCS loop ΔT variable input ≤ 50.7% RTP and TTD=0 for RCS loop ΔT variable input > 50.7 % RTP	TTD ≤ TD (Note 3) for RCS loop ΔT variable input 50% RTP TTD=0 for RCS loop ΔT variable input 50% RTP
15. Not used						

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.
(l) Minimum measured flow (MMF) is 89,800 gpm per loop for Unit 1 and 90,625 gpm per loop for Unit 2.

3.3.1-1

Table ~~3.3.3-1~~ (page 4 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
16. Turbine Trip						
a. Low Auto-Stop Oil Pressure	1[j]	3	O	SR 3.3.1.10 ~~SR 3.3.1.15~~	≥ 46.5 psig	50 psig
b. Turbine Stop Valve Closure	1[j]	4	P	~~SR 3.3.1.10~~ SR 3.3.1.15	≥ 1% open	1% open
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.14	NA	NA
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2[e]	2	S	SR 3.3.1.11 SR 3.3.1.13	≥ 8E-11 amp	1E-10 amp
b Low Power Reactor Trips Block, P-7	1	1 per train	T	SR 3.3.1.5	NA	NA
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 36.2% RTP	35% RTP

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(j) Above the P-9 (Power Range Neutron Flux) interlock.

Table ~~3.3.3-1~~ 3.3.1-1 (page 5 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
18. Reactor Trip System Interlocks (cont)						
d. Power Range Neutron Flux, P-9	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 51.2% RTP	50% RTP
e. Power Range Neutron Flux, P-10	1,2	4	S	SR 3.3.1.11 SR 3.3.1.13	≥ 8.8% RTP and ≤ 11.2% RTP	10% RTP
f. Turbine Impulse Chamber Pressure, P-13	1	2	T	SR 3.3.1.10 SR 3.3.1.13	≤ 10.2% RTP turbine impulse pressure equivalent	10% RTP turbine impulse pressure equivalent
19. Reactor Trip Breakers[k] (RTBs)	1,2	2 trains	R	SR 3.3.1.4	NA	NA
	3[b], 4[b], 5[b]	2 trains	C	SR 3.3.1.4	NA	NA
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms [k]	1,2	1 each per RTB	U	SR 3.3.1.4	NA	NA
	3[b], 4[b], 5[b]	1 each per RTB	C	SR 3.3.1.4	NA	NA
21. Automatic Trip Logic	1,2	2 trains	Q	SR 3.3.1.5	NA	NA
	3[b], 4[b], 5[b]	2 trains	C	SR 3.3.1.5	NA	NA
22. Seismic Trip	1,2	3 directions (x,y,z) in 3 locations	W	SR 3.3.1.12 SR 3.3.1.14 ~~SR 3.3.1.17~~	≤ 0.43g SR 3.3.1.5	0.35g

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.

(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

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

3.3.1-1

Table ~~3.3.3-1~~ (page 6 of 7)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.46% of ΔT span for hot leg or cold leg temperature inputs, 0.14% ΔT span for pressurizer pressure input, 0.19% ΔT span for ΔI inputs.

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

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T' is the nominal loop specific indicated T_{avg} at RTP, ≤ 576.6 (Unit 1) & 577.6 (Unit 2)°F.

P is the measured pressurizer pressure, psig

P' is the nominal RCS operating pressure, = 2235 psig

$K_1 = 1.20$	$K_2 = 0.0182/°F$	$K_3 = 0.000831/psig$
$\tau_1 = 30$ sec	$\tau_2 = 4$ sec	
$\tau_4 = 0$ sec	$\tau_5 = 0$ sec	
$f_1(\Delta I) =$	$-0.0275\{19 + (q_t - q_b)\}$	when $q_t - q_b \leq -19\%$ RTP
	0% of RTP	when $-19\% \text{ RTP} < q_t - q_b \leq 7\%$ RTP
	$0.0238\{(q_t - q_b) - 7\}$	when $q_t - q_b > 7\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

3.3.1-1

Table ~~3.3.3-1~~ (page 7 of 7)

Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.46% of ΔT span for hot leg or cold leg temperature inputs.

$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T^0 \left\{ K_4 - K_5 \frac{\tau_3 s}{1+\tau_3 s} T - K_6 \left[T - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T'' is the nominal loop specific indicated T_{avg} at RTP, ≤ 576.6 (Unit 1) & 577.6 (Unit 2)°F.

K_4 = 1.072 K_5 = 0.0174/°F for increasing T_{avg} 0/°F for decreasing T_{avg} K_6 = 0.00145/°F when T > T'' 0/°F when T ≤ T''

τ_3 = 10 sec τ_4 = 0 sec τ_5 = 0 sec

$f_2(\Delta I)$ = 0% RTP for all ΔI.

Note 3: Steam Generator Water-Level Low Low Time Delay

The Steam Generator Water Level-Low Low time delay function power allowable value shall not exceed the following trip setpoint power by more than 0.7% RTP.

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

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

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

B1 = -0.007128 sec/$(RTP)^3$

B2 = +0.8099 sec/$(RTP)^2$

B3 = -31.40 sec/(RTP)

B4 = +464.1 sec

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. One channel inoperable.	N.1 Restore channel to OPERABLE status. OR N.2 Declare the associated AFW pump or MSIV inoperable.	48 hours Immediately
O. One channel inoperable	----------NOTE---------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ---------- O.1 Place channel in trip. OR O.2.1 Be in MODE 3 AND O.2.2 Be in MODE 5.	 6 hours 12 hours 42 hours
P. One channel inoperable.	----------NOTE---------- ~~The inoperable~~ channel may be bypassed for up to 4 hours for surveillance testing ~~of other channels~~. ---------- P.1 Place channel in bypass. OR P.2.1 Be in MODE 3 AND P.2.2 Be in MODE 5.	 6 hours 12 hours 42 hours

Table 3.3.2-1 (page 3 of 7)
Engineered Safety feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
3. Containment Isolation (continued)						
b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	2 per train	B	SR 3.3.2.8	NA	NA
(2) 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) Containment Pressure High-High	1,2,3,4	4	P	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 22.12 psig	22 psig
4. Steam Line Isolation						
a. Manual Initiation	1,2[(i)],3[(i)]	1/valve	N	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2[(i)],3[(i)]	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure-High -High	1,2[(i)],3[(i)]	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 22.12 psig	22.0 psig

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.

(i) Except when all MSIVs are closed and de-activated.

SURVEILLANCE REQUIREMENTS

---NOTE---

SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

	SURVEILLANCE	FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.3.2	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.3.3	Perform CHANNEL CALIBRATION for Hydrogen Monitors	92 days

NOTE
Neutron detectors are excluded from CHANNEL CALIBRATION

3.3 INSTRUMENTATION

3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

LCO 3.3.5 One channel per bus of loss of voltage DG start Function; and two channels per bus of degraded voltage Function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources-Shutdown."

ACTIONS

NOTE
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more channels per bus inoperable.	A.1 NOTE One channel may be bypassed for up to 2 hours for surveillance testing. Enter applicable Condition(s) and Required Action(s) for the associated DG made inoperable by LOP DG start instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Not used	18
SR 3.3.5.2	Perform TADOT.	~~24~~ months
SR 3.3.5.3	Perform CHANNEL CALIBRATION with Allowable Value setpoints as follows: a. Loss of voltage Diesel Start Allowable Value ≥ 0 V with a time delay of ≤ 0.8 seconds and ≥ 2583 V with a ≤ 10 second time delay. Loss of voltage initiation of load shed with one relay Allowable Value ≥ 0 V with a time delay of ≤ 4 seconds and ≥ 2583 V with a time delay ≤ 25 seconds and with one relay Allowable Value ≥ 2870 V, instantaneous.	~~24~~ 18 months (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two required loops inoperable. OR No RCS or RHR loop in operation.	B.1 Suspend all operations involving a reduction of RCS boron concentration. AND B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify SG secondary side water levels are ≥ 15% for required RCS loops.	12 hours
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	~~12 hours~~ 7days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops-MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

NOTES

1. All RHR pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:
   a. The core outlet temperature is maintained at least 10°F below saturation temperature.
   b. No operations are permitted that would cause a reduction of the RCS boron concentration; and
   c. No draining operations to further reduce the RCS water volume are permitted.
2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

NOTE

While this LCO is not met, entry into MODE 5, Loops Not Filled, from MODE 5, Loops Filled, is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR loop inoperable.	A.1	Initiate action to restore RHR loop to OPERABLE status.	Immediately
B. Required RHR loops inoperable. OR No RHR loop in operation.	B.1	Suspend all operations involving reduction in RCS boron concentration.	Immediately
	AND		
	B.2	Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

~~d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and~~

d. 150 gallons per day primary to secondary LEAKAGE through any one SG.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

* For MODES 3 and 4, if steam generator water samples indicate less than the minimum detectable activity of 5.0 E-7 microcuries/ml for principal gamma emitters, the leakage requirement of specification 3.4.13.d may be considered met.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve. OR	72 hours
	A.2.2 Restore RCS PIV to within limits.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3. AND	6 hours
	B.2 Be in MODE 5.	36 hours

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, and 24 months AND

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1 (continued)		Within 24 hours following valve actuation due to automatic or manual action or flow through the valve except for valves 8802A, 8802B, and 8703 (8701, 8702)
SR 3.4.14.2	Not used	
SR 3.4.14.3	Not used	

Table 3.7.1-1 (page 1 of 1)
Maximum Allowable Power Range Neutron Flux High Setpoint With Inoperable MSSVs

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT %RTP
4	87 *
3	47 *
2	29 *

* Unless the Reactor Trip System breakers are in the open position.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two CRVS trains inoperable in MODE 5 OR 6, or during movement of irradiated fuel assemblies.	D.1 Suspend CORE ALTERATIONS. AND D.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
E. Two CRVS trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRVS train for ≥ 10 continuous hours with the heaters operating.	31 days
SR ~~3.17.10.2~~ 3.7.10.2	Verify that each CRVS redundant fan is aligned to receive electrical power from a separate OPERABLE vital bus.	31 days
SR 3.7.10.3	Perform required CRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.4	Verify each CRVS train automatically switches into the pressurization mode of operation on an actual or simulated actuation signal.	24 months
SR 3.7.10.5	Verify one CRVS train can maintain a positive pressure of ≥ 0.125 inches water gauge, relative to the outside atmosphere during the pressurization mode of operation.	24 months on a STAGGERED TEST BASIS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each FHBVS train for ≥ 15 minutes.	31 days
SR 3.7.13.2	Perform required FHBVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each FHBVS train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.13.4	Verify one FHBVS train can maintain a pressure ≤ -0.125 inches water gauge with respect to atmospheric pressure during the post accident mode of operation.	24 months on a STAGGERED TEST BASIS
SR 3.7.13.5	Verify damper M-29 can be closed.	24 months

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Vice President, Diablo Canyon Operations and Plant Manager, hereafter called Plant Manager, shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The Plant Manager or his designee, hereafter called Plant Manager, shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

5.1.2 The Shift Foreman (SFM) shall be responsible for the control room command function. During any absence of the SFM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual ~~(other than the Shift Technical Advisor)~~ with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SFM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.5 Programs and Manuals

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

2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.5.9-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.c.1. The interval may then be extended to a maximum of once per 40 months; and

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

   a) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13; or

   b) A seismic occurrence greater than the Double Design Earthquake, or

   c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   d) A main steam line or feedwater line break.

d. Acceptance Criteria

1. As used in this Specification:

   a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

   b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

   c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

   d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

   e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

   f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

      1) This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 5.5.9.d.1.j for the repair limit applicable to these intersections.

(continued)

5.5 Programs and Manuals

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

2) Axial cracks in tubes returned to service using W* shall have the upper crack tip below the BWT by at least the NDE measurement uncertainty, and below the top of tube sheet (TTS) by at least the NDE measurement uncertainty and crack growth allowance, such that at the end of the subsequent operating cycle the entire crack remains below the tubesheet secondary face.

3) Resolvable, single axial indications (multiple indications must return to the null point between individual cracks) within the flexible W* length can be left in service. Alternate RPC coils or an ultrasonic test (UT) inspection can be used to demonstrate return to null point between multiple axial indications or the absence of circumferential involvement between axial indications.

4) Tubes with inclined axial indications less than 2.0 inches long (including the crack growth allowance) having inclination angles relative to the tube axis of < 45 degrees minus the NDE uncertainty, ΔNDE_{CA}, on the measurement of the crack angle can be left in service. Tubes with two or more parallel (overlapping elevation), inclined axial cracks shall be plugged or repaired. For application of the 2.0 inch limit, an inclined indication is an axial crack that is visually inclined on the RCP C-scan, such that an angular measurement is required, and the measured angle exceeds the measurement uncertainty of ΔNDE_{CA}.

5) Circumferential, volumetric, and axial indications with inclination angles greater than (45 degrees - ΔNDE_{CA}) within the flexible W* length shall be plugged or repaired.

6) Any type of combination of the tube degradation below the W* length is acceptable.

2. The SG tube integrity shall be determined after completing the corresponding actions (plug all tubes exceeding the plugging limit ~~and all tubes containing through-wall cracks~~) required by Table 5.5.9-2.

e. Reports

The contents and frequency of reports concerning the SG tube surveillance program shall be in accordance with Specification 5.6.10.

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified below and in accordance with Regulatory Guide 1.52, Revision 2, ANSI N510 1980, and ASTM D3803-1989.

a. Demonstrate for each of the ESF systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per ~~operating cycle~~ 24 months

ESF Ventilation System	Flowrate
Control Room	2100 cfm
Auxiliary Building	73,500 cfm
Fuel Handling Building	35,750 cfm

b. Demonstrate for each of the ESF systems that an in-place test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per operating cycle.

ESF Ventilation System	Flowrate
Control Room	2100 cfm
Auxiliary Building	73,500 cfm
Fuel Handling Building	35,750 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal absorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and at the relative humidity specified below. Laboratory testing shall be completed at least once per 18 months and after every 720 hours of charcoal operation.

ESF Ventilation System	Penetration	RH
Control Room	1.0%	70%
Auxiliary Building	6.0%	70%
Fuel Handling Building	4.3%	95%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per ~~operating cycle~~ 24 MONTHS.

ESF Ventilation System	Delta P	Flowrate
Control Room	3.5 in. WG	2100 cfm
Auxiliary Building	3.7 in. WG	73,500 cfm
Fuel Handling Building	4.1 in. WG	35,750 cfm

e. Demonstrate that the charcoal pre-heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI N510-1980 at least once per ~~operating cycle~~ 24 MONTHS

ESF Ventilation System	Wattage
Control Room	5 ± 1 kW
Auxiliary Building	50 ± 5 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in temporary unprotected outdoor liquid storage tanks.

The gaseous radioactivity quantities shall be determined following the methodology in Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure." The liquid radwaste quantities shall be maintained such that 10 CFR Part 20 limits are met.

(continued)

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

or personnel continuously escorted by such individuals

e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift ~~supervisor~~ manager radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

(continued)

REVISED TECHNICAL SPECIFICATIONS PAGES

1.2 Logical Connectors

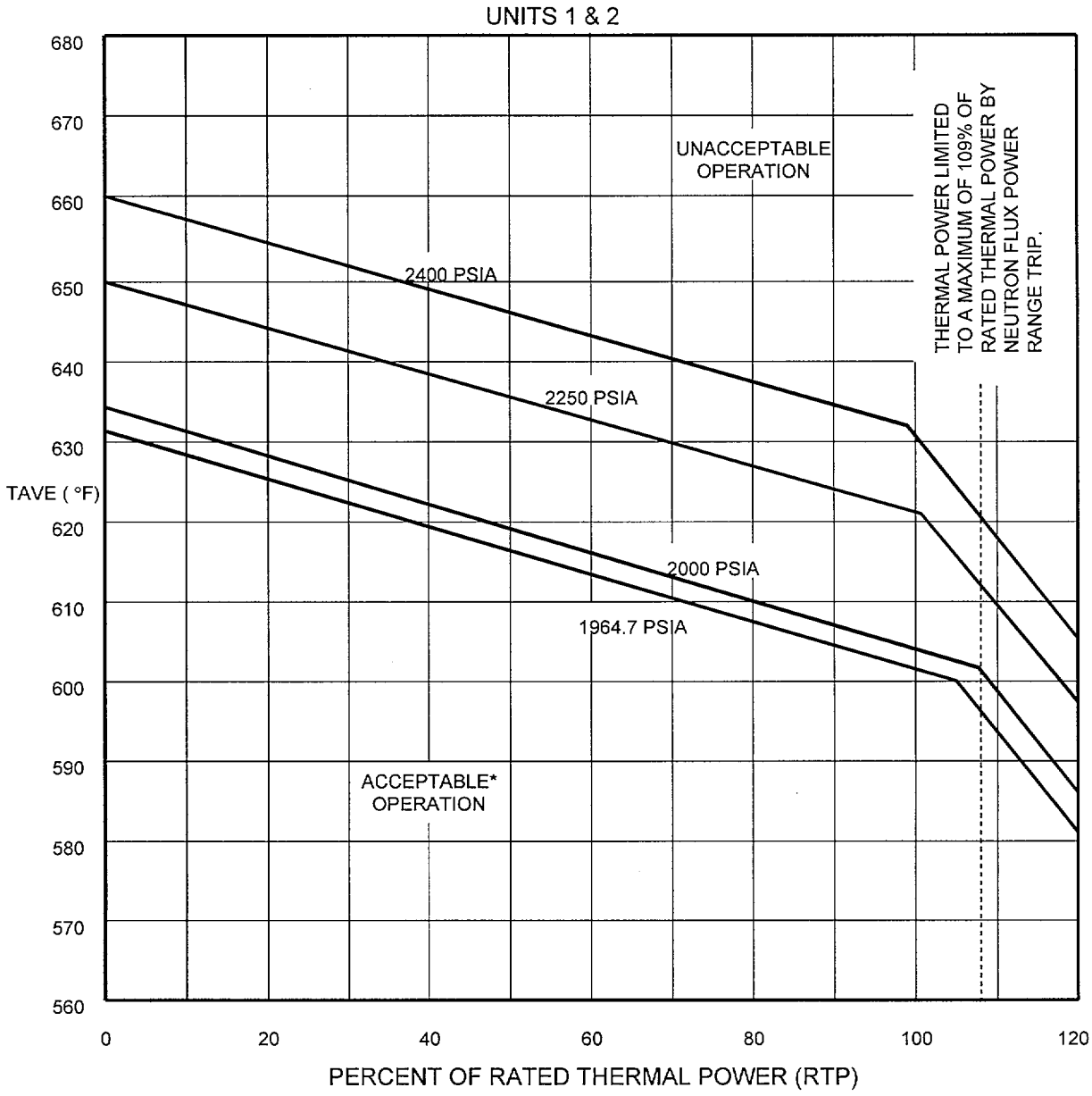
EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

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

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



*WHEN OPERATING IN THE REDUCED RTP REGION OF TECHNICAL SPECIFICATION LCO 3.4.1 (TABLE 3.4.1-1 FOR UNIT 1 AND TABLE 3.4.1-2 FOR UNIT 2) THE RESTRICTED POWER LEVEL MUST BE CONSIDERED 100% RTP FOR THIS FIGURE.

Figure 2.1.1-1

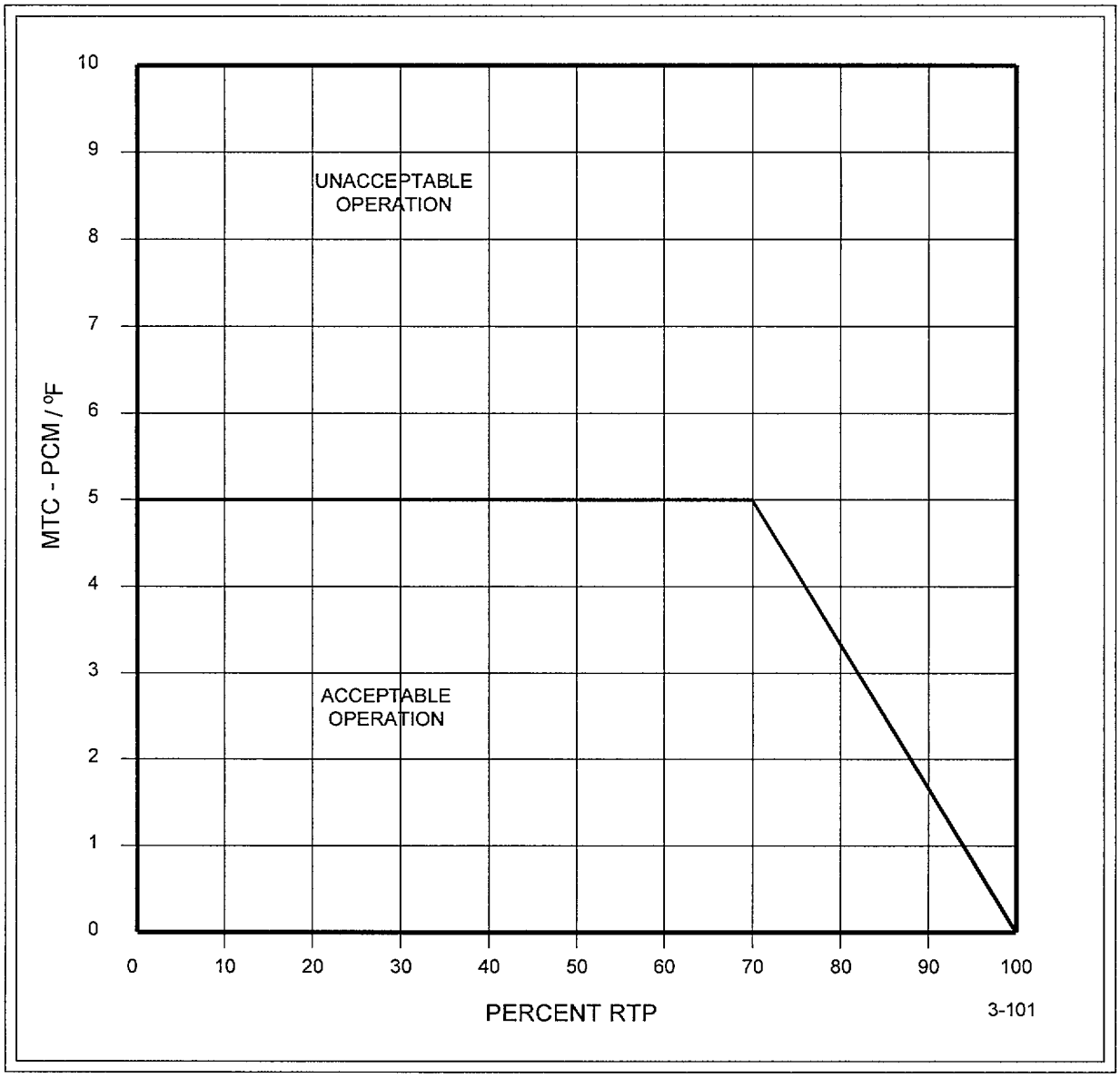


Figure 3.1.3-1 (page 1 of 1)

MODERATOR TEMPERATURE COEFFICENT vs. POWER LEVEL

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ---------- $F^N_{\Delta H}$ not within limit.	A.1.1	Restore $F^N_{\Delta H}$ within limit.	4 hours
	OR		
	A.1.2.1	Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND		
	A.1.2.2	Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP.	72 hours
	AND		
	A.2	Perform SR 3.2.2.1.	24 hours
	AND		(continued)

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s) or trains.	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status. OR B.2 Be in MODE 3.	48 hours 54 hours
----------NOTE---------- While this LCO is not met for function 19, 20 or 21, in MODE 5, making the Rod Control System capable of rod withdrawal is not permitted. ----------		
C. One channel or train inoperable.	C.1 Restore channel or train to OPERABLE status. OR C.2.1 Initiate action to fully insert all rods.	48 hours 48 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One channel inoperable.	----------NOTE---------- The inoperable channel, or one additional channel for functions 6, 7, and 8.b may be bypassed for up to 4 hours for surveillance testing of other channels. For functions 2.b, 3.a, 3.b, and 14.a only the inoperable channel may be bypassed for surveillance testing of other channels. ---------- E.1 Place channel in trip. OR E.2 Be in MODE 3.	 6 hours 12 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6. OR F.2 Increase THERMAL POWER to > P-10.	24 hours 24 hours
G. Two Intermediate Range Neutron Flux channels inoperable.	G.1 Suspend operations involving positive reactivity additions. AND G.2 Reduce THERMAL POWER to < P-6.	Immediately 2 hours
H. Not used		
I. One Source Range Neutron Flux channel inoperable.	I.1 Suspend operations involving positive reactivity additions.	Immediately
J. Two Source Range Neutron Flux channels inoperable.	J.1 Open reactor trip breakers (RTBs).	Immediately

(continued)

Table 3.3.1-1 (page 1 of 7)

Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA	NA
	3[b], 4[b], 5[b]	2	C	SR 3.3.1.14	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 110.2% RTP	109% RTP
b. Low	1[c],2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 26.2% RTP	25% RTP
3. Power Range Neutron Flux Rate						
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 5.6% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 5.6% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1[c], 2[d]	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 30.6% RTP	25% RTP

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlocks.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

Table 3.3.1-1 (page 2 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
5. Source Range Neutron Flux	2[e]	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 1.4 E5 cps	1.0 E5 cps
	3[b], 4[b], 5[b]	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 1.4 E5 cps	1.0 E5 cps
	3[f], 4[f], 5[f]	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	N/A
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1 (Page 3.3-17)	Refer to Note 1 (Page 3.3-17)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 2 (Page 3.3-18)	Refer to Note 2 (Page 3.3-18)
8. Pressurizer Pressure						
a. Low	1[g]	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1947.5 psig	1950 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2387.5 psig	2385 psig
9. Pressurizer Water Level—High	1[g]	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 90.2%	90%

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(f) With the RTBs open or all rods fully inserted and incapable of withdrawal. In this condition, source range Function does not provide reactor trip but does provide indication.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 3 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
10. Reactor Coolant Flow—Low	1[g]	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 89.8%[l] of MMF/ loop	90%[l] of MMF/loop
11. Reactor Coolant Pump (RCP) Breaker Position	1[g]	1 per RCP	M	SR 3.3.1.14	NA	NA
12. Undervoltage RCPs	1[g]	2 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 7877 V each bus	8050 V each bus
13. Underfrequency RCPs	1[g]	3 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 53.9 Hz each bus	54.0 Hz each bus
14. a. Steam Generator (SG) Water Level—Low Low	1,2	3 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 7.0%	7.2%
b. SG Water Level - Low Low Trip Time Delay (TTD)	1,2	4	X	SR 3.3.1.7 SR 3.3.1.10	TTD ≤ 1.01 TD (Note 3) for RCS loop ΔT variable input ≤ 50.7% RTP and TTD=0 for RCS loop ΔT variable input > 50.7 % RTP	TTD ≤ TD (Note 3) for RCS loop ΔT variable input 50% RTP TTD=0 for RCS loop ΔT variable input 50% RTP
15. Not used						

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.
(l) Minimum measured flow (MMF) is 89,800 gpm per loop for Unit 1 and 90,625 gpm per loop for Unit 2.

Table 3.3.1-1 (page 4 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
16. Turbine Trip						
a. Low Auto-Stop Oil Pressure	1[(j)]	3	O	SR 3.3.1.10	≥ 46.5 psig	50 psig
b. Turbine Stop Valve Closure	1[(j)]	4	P	SR 3.3.1.15	≥ 1% open	1% open
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.14	NA	NA
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2[(e)]	2	S	SR 3.3.1.11 SR 3.3.1.13	≥ 8E-11 amp	1E-10 amp
b Low Power Reactor Trips Block, P-7	1	1 per train	T	SR 3.3.1.5	NA	NA
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 36.2% RTP	35% RTP

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.
(j) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 5 of 7)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
18. Reactor Trip System Interlocks (cont)						
d. Power Range Neutron Flux, P-9	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 51.2% RTP	50% RTP
e. Power Range Neutron Flux, P-10	1,2	4	S	SR 3.3.1.11 SR 3.3.1.13	≥ 8.8% RTP and ≤ 11.2% RTP	10% RTP
f. Turbine Impulse Chamber Pressure, P-13	1	2	T	SR 3.3.1.10 SR 3.3.1.13	≤ 10.2% RTP turbine impulse pressure equivalent	10% RTP turbine impulse pressure equivalent
19. Reactor Trip Breakers[k] (RTBs)	1,2	2 trains	R	SR 3.3.1.4	NA	NA
	3[b], 4[b], 5[b]	2 trains	C	SR 3.3.1.4	NA	NA
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms [k]	1,2	1 each per RTB	U	SR 3.3.1.4	NA	NA
	3[b], 4[b], 5[b]	1 each per RTB	C	SR 3.3.1.4	NA	NA
21. Automatic Trip Logic	1,2	2 trains	Q	SR 3.3.1.5	NA	NA
	3[b], 4[b], 5[b]	2 trains	C	SR 3.3.1.5	NA	NA
22. Seismic Trip	1,2	3 directions (x,y,z) in 3 locations	W	SR 3.3.1.5 SR 3.3.1.12 SR 3.3.1.14	≤ 0.43g	0.35g

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.

(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

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

Table 3.3.1-1 (page 6 of 7)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.46% of ΔT span for hot leg or cold leg temperature inputs, 0.14% ΔT span for pressurizer pressure input, 0.19% ΔT span for ΔI inputs.

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

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T' is the nominal loop specific indicated T_{avg} at RTP, ≤ 576.6 (Unit 1) & 577.6 (Unit 2)°F.

P is the measured pressurizer pressure, psig

P' is the nominal RCS operating pressure, = 2235 psig

K_1 = 1.20	K_2 = 0.0182/°F	K_3 = 0.000831/psig
τ_1 = 30 sec	τ_2 = 4 sec	
τ_4 = 0 sec	τ_5 = 0 sec	
$f_1(\Delta I)$ =	$-0.0275\{19 + (q_t - q_b)\}$	when $q_t - q_b \leq -19\%$ RTP
	0% of RTP	when $-19\% \text{ RTP} < q_t - q_b \leq 7\%$ RTP
	$0.0238\{(q_t - q_b) - 7\}$	when $q_t - q_b > 7\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 7 of 7)

Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.46% of ΔT span for hot leg or cold leg temperature inputs.

$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T^0 \left\{ K_4 - K_5 \frac{\tau_3 s}{1+\tau_3 s} T - K_6 \left[T - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T" is the nominal loop specific indicated T_{avg} at RTP, ≤ 576.6 (Unit 1) & 577.6 (Unit 2)°F.

K_4 = 1.072 K_5 = 0.0174/°F for increasing T_{avg} 0/°F for decreasing T_{avg} K_6 = 0.00145/°F when T > T" 0/°F when T ≤ T"

τ_3 = 10 sec τ_4 = 0 sec τ_5 = 0 sec

$f_2(\Delta I)$ = 0% RTP for all ΔI.

Note 3: Steam Generator Water-Level Low Low Time Delay

The Steam Generator Water Level-Low Low time delay function power allowable value shall not exceed the following trip setpoint power by more than 0.7% RTP.

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

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

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

B1 = -0.007128 $sec/(RTP)^3$

B2 = +0.8099 $sec/(RTP)^2$

B3 = -31.40 sec/(RTP)

B4 = +464.1 sec

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. One channel inoperable.	N.1 Restore channel to OPERABLE status. OR N.2 Declare the associated AFW pump or MSIV inoperable.	48 hours Immediately
O. One channel inoperable	----------NOTE---------- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ---------- O.1 Place channel in trip. OR O.2.1 Be in MODE 3 AND O.2.2 Be in MODE 5.	 6 hours 12 hours 42 hours
P. One channel inoperable.	----------NOTE---------- One additional channel may be bypassed for up to 4 hours for surveillance testing. ---------- P.1 Place channel in bypass. OR P.2.1 Be in MODE 3 AND P.2.2 Be in MODE 5.	 6 hours 12 hours 42 hours

Table 3.3.2-1 (page 3 of 7)
Engineered Safety feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL[a] TRIP SETPOINT
3. Containment Isolation (continued)						
b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	2 per train	B	SR 3.3.2.8	NA	NA
(2) 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) Containment Pressure High-High	1,2,3,4	4	P	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≤ 22.12 psig	22 psig
4. Steam Line Isolation						
a. Manual Initiation	1,2[(i)],3[(i)]	1/valve	N	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2[(i)],3[(i)]	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure-High -High	1,2[(i)],3[(i)]	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 22.12 psig	22.0 psig

(continued)

(a) A channel is OPERABLE with an actual Trip Setpoint value outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint. A Trip Setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions.

(i) Except when all MSIVs are closed and de-activated.

SURVEILLANCE REQUIREMENTS

----------NOTE----------

SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

	SURVEILLANCE	FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.3.2	----------Note---------- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.3.3	Perform CHANNEL CALIBRATION for Hydrogen Monitors	92 days

3.3 INSTRUMENTATION

3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

LCO 3.3.5 One channel per bus of loss of voltage DG start Function; and two channels per bus of degraded voltage Function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated DG is required to be OPERABLE by LCO 3.8.2, "AC Sources-Shutdown."

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more channels per bus inoperable.	A.1 ----------NOTE---------- One channel may be bypassed for up to 2 hours for surveillance testing. ---------- Enter applicable Condition(s) and Required Action(s) for the associated DG made inoperable by LOP DG start instrumentation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Not used	
SR 3.3.5.2	Perform TADOT.	18 months
SR 3.3.5.3	Perform CHANNEL CALIBRATION with Allowable Value setpoints as follows: a. Loss of voltage Diesel Start Allowable Value ≥ 0 V with a time delay of ≤ 0.8 seconds and ≥ 2583 V with a ≤ 10 second time delay. Loss of voltage initiation of load shed with one relay Allowable Value ≥ 0 V with a time delay of ≤ 4 seconds and ≥ 2583 V with a time delay ≤ 25 seconds and with one relay Allowable Value ≥ 2870 V, instantaneous.	18 months

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two required loops inoperable. OR No RCS or RHR loop in operation.	B.1 Suspend all operations involving a reduction of RCS boron concentration. AND B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify SG secondary side water levels are ≥ 15% for required RCS loops.	12 hours
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops-MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

----------NOTES----------

1. All RHR pumps may be removed from operation for ≤ 1 hour per 8 hour period provided:
   a. The core outlet temperature is maintained at least 10°F below saturation temperature.
   b. No operations are permitted that would cause a reduction of the RCS boron concentration; and
   c. No draining operations to further reduce the RCS water volume are permitted.
2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

----------NOTE----------

While this LCO is not met, entry into MODE 5, Loops Not Filled, from MODE 5, Loops Filled, is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately
B. Required RHR loops inoperable. OR No RHR loop in operation.	B.1 Suspend all operations involving reduction in RCS boron concentration. AND B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 150 gallons per day primary to secondary LEAKAGE through any one SG.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

* For MODES 3 and 4, if steam generator water samples indicate less than the minimum detectable activity of 5.0 E-7 microcuries/ml for principal gamma emitters, the leakage requirement of specification 3.4.13.d. may be considered met.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve. OR	72 hours
	A.2.2 Restore RCS PIV to within limits.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3. AND	6 hours
	B.2 Be in MODE 5.	36 hours

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, and 24 months AND (continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1 (continued)		Within 24 hours following valve actuation due to automatic or manual action or flow through the valve except for valves 8701, 8702, 8802A, 8802B, and 8703
SR 3.4.14.2	Not used	
SR 3.4.14.3	Not used	

Table 3.7.1-1 (page 1 of 1)
Maximum Allowable Power Range Neutron Flux High Setpoint With Inoperable MSSVs

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT %RTP
4	87 *
3	47 *
2	29 *

* Unless the Reactor Trip System breakers are in the open position.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two CRVS trains inoperable in MODE 5 OR 6, or during movement of irradiated fuel assemblies.	D.1 Suspend CORE ALTERATIONS. AND D.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
E. Two CRVS trains inoperable in MODE 1, 2, 3, or 4.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRVS train for ≥ 10 continuous hours with the heaters operating.	31 days
SR 3.7.10.2	Verify that each CRVS redundant fan is aligned to receive electrical power from a separate OPERABLE vital bus.	31 days
SR 3.7.10.3	Perform required CRVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.4	Verify each CRVS train automatically switches into the pressurization mode of operation on an actual or simulated actuation signal.	24 months
SR 3.7.10.5	Verify one CRVS train can maintain a positive pressure of ≥ 0.125 inches water gauge, relative to the outside atmosphere during the pressurization mode of operation.	24 months on a STAGGERED TEST BASIS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each FHBVS train for ≥ 15 minutes.	31 days
SR 3.7.13.2	Perform required FHBVS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each FHBVS train actuates on an actual or simulated actuation signal.	24 months
SR 3.7.13.4	Verify one FHBVS train can maintain a pressure ≤ -0.125 inches water gauge with respect to atmospheric pressure during the post accident mode of operation.	24 months on a STAGGERED TEST BASIS
SR 3.7.13.5	Verify damper M-29 can be closed.	24 months

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 The Vice President, Diablo Canyon Operations and Plant Manager, hereafter called Plant Manager, shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The Plant Manager or his designee, hereafter called Plant Manager, shall approve, prior to implementation, each proposed test, experiment, or modification to systems or equipment that affect nuclear safety.

5.1.2 The Shift Foreman (SFM) shall be responsible for the control room command function. During any absence of the SFM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SFM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

5.5 Programs and Manuals

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

2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.5.9-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.c.1. The interval may then be extended to a maximum of once per 40 months; and

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

a) Reactor-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13; or

b) A seismic occurrence greater than the Double Design Earthquake, or

c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

d) A main steam line or feedwater line break.

d. Acceptance Criteria

1. As used in this Specification:

a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

1) This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 5.5.9.d.1.j for the repair limit applicable to these intersections.

(continued)

5.5 Programs and Manuals

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

2) Axial cracks in tubes returned to service using W* shall have the upper crack tip below the BWT by at least the NDE measurement uncertainty, and below the top of tube sheet (TTS) by at least the NDE measurement uncertainty and crack growth allowance, such that at the end of the subsequent operating cycle the entire crack remains below the tubesheet secondary face.

3) Resolvable, single axial indications (multiple indications must return to the null point between individual cracks) within the flexible W* length can be left in service. Alternate RPC coils or an ultrasonic test (UT) inspection can be used to demonstrate return to null point between multiple axial indications or the absence of circumferential involvement between axial indications.

4) Tubes with inclined axial indications less than 2.0 inches long (including the crack growth allowance) having inclination angles relative to the tube axis of < 45 degrees minus the NDE uncertainty, ΔNDE_{CA}, on the measurement of the crack angle can be left in service. Tubes with two or more parallel (overlapping elevation), inclined axial cracks shall be plugged or repaired. For application of the 2.0 inch limit, an inclined indication is an axial crack that is visually inclined on the RCP C-scan, such that an angular measurement is required, and the measured angle exceeds the measurement uncertainty of ΔNDE_{CA}.

5) Circumferential, volumetric, and axial indications with inclination angles greater than (45 degrees - ΔNDE_{CA}) within the flexible W* length shall be plugged or repaired.

6) Any type of combination of the tube degradation below the W* length is acceptable.

2. The SG tube integrity shall be determined after completing the corresponding actions (plug all tubes exceeding the plugging) required by Table 5.5.9-2.

e. Reports

The contents and frequency of reports concerning the SG tube surveillance program shall be in accordance with Specification 5.6.10.

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified below and in accordance with Regulatory Guide 1.52, Revision 2, ANSI N510 1980, and ASTM D3803-1989.

a. Demonstrate for each of the ESF systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per 24 months.

ESF Ventilation System	Flowrate
Control Room	2100 cfm
Auxiliary Building	73,500 cfm
Fuel Handling Building	35,750 cfm

b. Demonstrate for each of the ESF systems that an in-place test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per operating cycle.

ESF Ventilation System	Flowrate
Control Room	2100 cfm
Auxiliary Building	73,500 cfm
Fuel Handling Building	35,750 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal absorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and at the relative humidity specified below. Laboratory testing shall be completed at least once per 18 months and after every 720 hours of charcoal operation.

ESF Ventilation System	Penetration	RH
Control Room	1.0%	70%
Auxiliary Building	6.0%	70%
Fuel Handling Building	4.3%	95%

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested in accordance with ANSI N510-1980 at the system flowrate specified below ± 10% at least once per 24 months.

ESF Ventilation System	Delta P	Flowrate
Control Room	3.5 in. WG	2100 cfm
Auxiliary Building	3.7 in. WG	73,500 cfm
Fuel Handling Building	4.1 in. WG	35,750 cfm

e. Demonstrate that the charcoal pre-heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI N510-1980 at least once per 24 months.

ESF Ventilation System	Wattage
Control Room	5 ± 1 kW
Auxiliary Building	50 ± 5 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in temporary unprotected outdoor liquid storage tanks.

The gaseous radioactivity quantities shall be determined following the methodology in Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure." The liquid radwaste quantities shall be maintained such that 10 CFR Part 20 limits are met.

(continued)

5.7 High Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation (continued)

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift manager, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

(continued)