DESCRIPTION OF PROPOSED CHANGES NPF-10-8 AND NPF-15-8
AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.6.3 - "Containment Isolation Valves.

Existing Specifications:

Unit 2: See Attachment "A"
Unit 3: See Attachment "C"

Proposed Specifications:

Unit 2: See Attachment "B"
Unit 3: See Attachment "D"

DESCRIPTION

LCO 3.6.3, its associated Action Statements, Surveillance Requirements and Table 3.6-1 have been revised to clarify determination of valve operability and the Actions to be taken should a valve become inoperable. Sections A, B and C of Table 3.6-1 have been revised to include only those valves assumed to close or be closed under post accident conditions while Section D of Table 3.6-1 now contains only those valves assumed or required to open or be opened under post accident conditions as stated in FSAR Table 6.2-35. The Action Statements and Surveillance Requirements for all valves listed have been clarified to be consistent with the accident analysis and other requirements of these Technical Specifications. The following describes each change and its justification.

1) LCO 3.6.3 itself has been revised to delete reference to "isolation time".
REASON: - Reference to "isolation time" in the LCO indicates that the LCO is only applicable to those valves for which an "isolation time" is specified in Table 3.6-1. By Specification 4.0.3 (and its basis) OPERABILITY is determined by satisfactory completion of Surveillance Requirements.

2) The Action statements have been modified to indicate that the existing requirements are only applicable to those valves assumed to close or be closed under post accident conditions and not to those required to be opened post accident. An Action statement pertaining to those valves assumed or required to be opened to mitigate the consequences of a design basis accident has been added.
REASON: Applying the existing Action statements to all valves listed in Table 3.6-1 places the unit in an unsafe condition if applied to those valves required to be opened post accident. An Action statement refering to the Action requirements of appropriate Engineered Safety Features LCO's has been added to recognize such valves.

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3) Surveillance Requirements 4.6.3.1, 4.6.3.2, and 4.6.3.3 have been revised to indicate their applicability to automatic isolation valves only. Surveillance Requirement 4.6.3.2 has been revised to indicate the automatic valves are tested with ESFAS test signals.
REASON: Each of these requirements make reference to "isolation time" or "automatic actuation" indicating these were intended for automatic valves only. The automatic valves are actuated by between one and three types of actuation signal depending on the valve's function. Specifying ESFAS test signals clarifies the Surveillance Requirement.

4) Surveillance Requirements 4.6.3.4 and 4.6.3.5 have been added to clarify how valves in Section C and D of Table 3.6-1 are determined to be OPERABLE.
REASON: As presently written, Surveillance Requirements do not clearly state how MANUAL and OTHER valves listed in Table 3.6-1 are determined to be OPERABLE.

5) Table 3.6-1 has been revised as follows:

SECTION A This section of the table has been revised to include only those automatic containment isolation valves designed in accordance with General Design Criteria (GDC) 54 through 57 of 10 CFR 50 Appendix A, and are relied upon in the accident analysis to close or be closed automatically for containment isolation purposes. MSIV's and MFIV's have been removed and CCW valves have been added.
REASON: GDC 54 through 57 are not applicable to MSIV's and MFIV's as indicated in FSAR Table 6.2-35. CCW valves HV-6223 and HV-6236 have been modified by approved design change to now receive a CIAS actuation. Containment minipurge valves HV-9821, 9823, 9824 and 9825 have been removed as they are also listed in SECTION B.

SECTION B

No changes have been made to SECTION B valves.

SECTION C

This section of the table has been revised to include only those manually operated or check valves assumed or required to be closed under design basis accident conditions and for which the General Design Criteria of 10 CFR 50 Appendix A is applicable as indicated in FSAR Table 6.2-35. Valves associated with secondary systems have been removed as have those manually operated valves assumed or required to be opened post accident. The asterisk footnote (*) has been applied to all valves in Section C.
REASON: These are the valves intended by Surveillance Requirement 4.6.1.1.a to be verified closed at least once every 31 days except when opened under Administrative control. The footnote has been added to permit periodic maintenance and testing under controlled conditions.

SECTION D

This section of the table has been revised to include only those valves assumed or required to be opened under design basis accident conditions and for which application of the existing Action statements would reduce the margin of safety provided by Engineered Safety Features.
REASON: For consistency with the intentions of this LCO.

The BASES for these Technical Specifications has been revised to clarify requirements.

Safety Analysis:

The proposed changes discussed above shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with these proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   1) Clarifying LCO 3.6.3 to remove reference to "isolation times" is considered editorial and does not affect the probability or consequences of an accident.

   2) If the Action statements, as presently written, are applied to valves assumed or required to be opened, the probability or consequences of an accident would increase. As revised, the Action to be taken for INOPERABLE valves is dependent upon their intended safety function, thereby maintaining the probabilities assumed in the safety analysis.

   3) Clarifying existing surveillance requirements is considered editorial and does not affect the probability or consequences of an accident.

   4) Adding Surveillance Requirements to clarify OPERABILITY requirements of manual and other valves is editorial in nature and does not affect the probability or consequences of an accident.

   5) Relocating valves within the Table to be consistent with revisions to the LCO, Action Statement and Surveillance Requirement is editorial in nature. Deletion of those valves for which GDC 54 through 57 provisions of 10 CFR 50 Appendix A are not applicable is consistent with FSAR Table 6.2-35 and do not therefore increase the probability or consequences of an accident previously evaluated. The MSIV's are subject to other LCO's within Technical Specification.

2. Will operation of the facility in accordance with these proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Operation of the facility in accordance with these proposed changes is not unlike operation under existing specifications. Revisions have been made to clarify this specification and is consistent with the evaluations currently presented in the FSAR.

3. Will operation of the facility in accordance with these proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The margin of safety assumed in the FSAR will be maintained by operation of the facility in accordance with these proposed changes.

<u>Safety and Significant Hazards Consideration Determination</u>

Based on the Safety Evaluation, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Environmental Statement.

Attachment "A"

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a CIAS or SIAS test signal, each isolation valve actuates to its isolation position.

SURVEILLANCE REQUIREMENTS (Continued)

b. Verifying that on a Containment Radiation-High test signal, all containment purge valves actuate to their isolation position.

4.6.3.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
A. CONTAINMENT ISOLATION (CIAS)			
1	HV-0510	Pressurizer steam space sample	40
1	HV-0511	Pressurizer steam space sample	40
2	TV-9267	Letdown line to letdown heat exchanger	40
2	HV-9205	Letdown line to letdown heat exchanger	40
4	HV-0508	Reactor coolant loops hot leg sample	40
4	HV-0509	Reactor coolant loops hot leg sample	40
4	HV-0517	Reactor coolant loops hot leg sample	40
6	HV-9334	Safety injection drain to RWST	40
7	HV-9217	Reactor coolant pump seal bleed off	40
7	HV-9218	Reactor coolant pump seal bleed off	40
11	HV-7911	Demineralized water to service station and sump pump	40
12	HV-0512	Pressurizer surge line sample	40
12	HV-0513	Pressurizer surge line sample	40
13	HV-5803	Containment sump pump discharge	40
13	HV-5804	Containment sump pump discharge	40
14	HV-5686	Fire protection	40
16C	HV-7805	Containment air radioactivity monitor inlet	1
16C	HV-7810	Containment air radioactivity monitor inlet	1
18	HV-9821	Containment minipurge inlet	5
18	HV-9823	Containment minipurge inlet	5
19	HV-9824	Containment minipurge outlet	5
19	HV-9825	Containment minipurge outlet	5
22	HV-5388	Instrument air supply line	40
23A	HV-5437	N_2 supply to quench tank, reactor coolant drain tank, and steam generators	40
26	HV-7512	Reactor coolant drain tank pump discharge	40
26	HV-7513	Reactor coolant drain tank pump discharge	40
27C	HV-7806	Containment air radioactivity monitor outlet	1
27C	HV-7811	Containment air radioactivity monitor outlet	1

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
28	HV-4052#	Steam generator feedwater	10
29	HV-4048#	Steam generator feedwater	10
30A	HV-7802	Containment air radioactivity monitor inlet	1
30A	HV-7803	Containment air radioactivity monitor inlet	1
30B	HV-7801	Containment air radioactivity monitor outlet	1
30B	HV-7800	Containment air radioactivity monitor outlet	1
30B	HV-7816	Containment air radioactivity monitor outlet	1
30C	HV-0516	Quench tank and drain tank gas sample	40
30C	HV-0514	Quench tank and drain tank gas sample	40
30C	HV-0515	Quench tank and drain tank gas sample	40
32	HV-8204#	Mainsteam isolation	5
33	HV-8205#	Mainsteam isolation	5
42	HV-6211	Component cooling water inlet	40
43	HV-6216	Component cooling water outlet	40
45	HV-9900	Containment normal A/C chilled water inlet	40
45	HV-9920	Containment normal A/C chilled water inlet	40
46	HV-9971	Containment normal A/C chilled water inlet	40
46	HV-9921	Containment normal A/C chilled water outlet	40
47	HV-7258	Containment waste gas vent header	40
47	HV-7259	Containment waste gas vent header	40
77	HV-5434	Nitrogen supply to safety injection tanks	40
B. CONTAINMENT PURGE (CPIS)			
18	HV-9949**	Containment purge inlet (normal)	12
18	HV-9948**	Containment purge inlet (normal)	12
18	HV-9821	Containment mini-purge inlet	5
18	HV-9823	Containment mini-purge inlet	5
19	HV-9950**	Containment purge outlet (normal)	12
19	HV-9951**	Containment purge outlet (normal)	12
19	HV-9824	Containment mini-purge outlet	5
19	HV-9825	Containment mini-purge outlet	5

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. MANUAL			
6	2"-099-C-334*	Safety injection drain to RWST	NA
8	HV-9200	Charging line to regenerative heat exchanger	NA
9	HV-9337#@	Shutdown cooling to LPSI pumps	NA
9	HV-9377#@	Shutdown cooling to LPSI pumps	NA
9	HV-9336#@	Shutdown cooling to LPSI pumps	NA
9	HV-9379#@	Shutdown cooling to LPSI pumps	NA
10A	HV-0352A#	Containment pressure detectors	NA
10C	3/4"-038-C-396	Integrated leak rate test pressure sensor	NA
10C	3/4"-039-C-396	Integrated leak rate test pressure sensor	NA
16A	HV-0500*	Post LOCA hydrogen monitor	NA
16A	HV-0501*	Post LOCA hydrogen monitor	NA
16B	HV-0502*	Post LOCA hydrogen monitor	NA
16B	HV-0503*	Post LOCA hydrogen monitor	NA
20	2"-321-C-376*	Quench tank makeup	NA
21	2"-055-C-387	Service air supply line	NA
25	10"-100-C-212	Refueling canal fill and drain	NA
25	10"-101-C-212	Refueling canal fill and drain	NA
27A	HV-0352D#	Containment pressure detectors	NA
31	HV-9946	Containment hydrogen purge inlet	NA
31	HCV-9945	Containment hydrogen purge inlet	NA
40A	HV-0352B#	Containment pressure detectors	NA
67	HV-9434	Hot leg injection	NA
68	2"-130-C-334	Charging line to auxiliary spray	NA
70	2"-037-C-387	Auxiliary steam inlet to utility stations	NA
70	2"-038-C-387	Auxiliary steam inlet to utility stations	NA
71	HV-9420	Hot leg injection	NA
73A	HV-0352C#	Containment pressure detectors	NA
74	HV-9917	Containment hydrogen purge outlet	NA
74	HCV-9918	Containment hydrogen purge outlet	NA

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
D. OTHER			
3	3"-018-A-551#	High pressure safety injection	NA
3	HV-9323#	High pressure safety injection	NA
3	HV-9324#	High pressure safety injection	NA
5	3"-019-A-551#	High pressure safety injection	NA
5	HV-9326#	High pressure safety injection	NA
5	HV-9327#	High pressure safety injection	NA
8	2"-122-C-554	Charging line to regenerative heat exchanger	NA
9	PSV-9349#	Shutdown cooling relief	NA
11	3"-236-C-675	Demineralized water to service stations and sump pump	NA
14	4"-061-C-681	Fire protection	NA
17	HV-4058#*	Steam generator secondary coolant sample	NA
20	2"-573-C-611	Quench tank makeup	NA
21	2"-017-C-627	Service air supply line	NA
22	1-1/2"-016-C-617	Instrument air supply line	NA
23A	3/4"-002-C-611	LP N_2 to containment	NA
32	HV-8421#	Mainsteam atmospheric dump	NA
32	PSV-8410#	Mainsteam relief	NA
32	PSV-8411#	Mainsteam relief	NA
32	PSV-8412#	Mainsteam relief	NA
32	PSV-8413#	Mainsteam relief	NA
32	PSV-8414#	Mainsteam relief	NA
32	PSV-8415#	Mainsteam relief	NA
32	PSV-8416#	Mainsteam relief	NA
32	PSV-8417#	Mainsteam relief	NA
32	PSV-8418#	Mainsteam relief	NA
32	HV-6249B#	Mainsteam trap isolation	NA
32	HV-8202#	Mainsteam isolation bypass	NA
32	HV-8200#	Mainsteam to auxiliary feedwater turbine	NA
33	HV-8419#	Mainsteam atmospheric dump	NA
33	PSV-8401#	Mainsteam relief	NA

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
33	PSV-8402#	Mainsteam relief	NA
33	PSV-8403#	Mainsteam relief	NA
33	PSV-8404#	Mainsteam relief	NA
33	PSV-8405#	Mainsteam relief	NA
33	PSV-8406#	Mainsteam relief	NA
33	PSV-8407#	Mainsteam relief	NA
33	PSV-8408#	Mainsteam relief	NA
33	PSV-8409#	Mainsteam relief	NA
33	HV-8248B#	Mainsteam trap isolation	NA
33	HV-8203#	Mainsteam isolation bypass	NA
33	HV-8201#	Mainsteam to auxiliary feedwater turbine	NA
36	HV-4054#*	Steam generator blowdown	NA
37	HV-4053#*	Steam generator blowdown	NA
39	3"-020-A-551#	High pressure safety injection	NA
39	HV-9329#	High pressure safety injection	NA
39	HV-9330#	High pressure safety injection	NA
41	3"-021-A-551#	High pressure safety injection	NA
41	HV-9332#	High pressure safety injection	NA
41	HV-9333#	High pressure safety injection	NA
42	HV-6223	Component cooling water inlet	NA
43	HV-6236	Component cooling water inlet	NA
44	HV-4057#*	Steam generator secondary coolant sample	NA
48	8"-072-A-552#@	Low pressure safety injection	NA
48	HV-9322#@	Low pressure safety injection	NA
49	8"-073-A-552#@	Low pressure safety injection	NA
49	HV-9325#@	Low pressure safety injection	NA
50	8"-074-A-552#@	Low pressure safety injection	NA
50	HV-9328#@	Low pressure safety injection	NA
51	8"-075-A-552#@	Low pressure safety injection	NA
51	HV-9331#@	Low pressure safety injection	NA
52	8"-004-C-406	Containment spray inlet	NA
52	HV-9367	Containment spray inlet	NA

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
53	8"-006-C-406	Containment spray inlet	NA
53	HV-9368	Containment spray inlet	NA
54	HV-9304#	Containment emergency sump recirculation	NA
54	HV-9302#	Containment emergency sump recirculation	NA
55	HV-9305#	Containment emergency sump recirculation	NA
55	HV-9303#	Containment emergency sump recirculation	NA
56	HV-6366	Containment emergency A/C cooling water inlet	NA
57	HV-6372	Containment emergency A/C cooling water inlet	NA
58	HV-6368	Containment emergency A/C cooling water inlet	NA
59	HV-6370	Containment emergency A/C cooling water inlet	NA
60	HV-6369	Containment emergency A/C cooling water inlet	NA
61	HV-6371	Containment emergency A/C cooling water inlet	NA
62	HV-6367	Containment emergency A/C cooling water inlet	NA
63	HV-6373	Containment emergency A/C cooling water inlet	NA
67	3"-157-A-551	Hot leg injection	NA
68	2"-129-A-554	Charging line to auxiliary spray	NA
71	3"-158-A-551	Hot leg injection	NA
75	HV-4715#	Steam generator auxiliary feedwater	NA
75	HV-4731#	Steam generator auxiliary feedwater	NA
77	2"-108-C-627	Nitrogen supply to safety injection tanks	NA
78	HV-4714#	Steam generator auxiliary feedwater	NA
78	HV-4730#	Steam generator auxiliary feedwater	NA

*May be opened on an intermittent basis under administrative control.

**Power to the valve removed in accordance with Specification 3.6.1.7.

#Not subject to Type C leakage tests.

@Shutdown cooling valves may be opened in MODE 4.

3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the iodine removal system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure that the solution recirculated within containment after a LOCA has a pH value between 8.0 and 10.0 at the end of the NaOH injection period. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

The 5 year Surveillance testing is intended to verify that no crystallization of the NaOH or other obstruction has occurred in the piping from the spray additive tank ot the suction of the containment spray pumps.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

The containment cooling system and the containment spray system are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the containment cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the containment spray system have been maintained consistent with that assigned other inoperable ESF equipment since the containment spray system also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

Attachment "B"

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

1. With one or more of the isolation valve(s) specified in Section A, B and C of Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration* that is open and either:

   a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

   b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured** in the isolation position, or one closed manual valve or blind flange, or

   c. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one or more of the valves specified in Section D of Table 3.6-1 inoperable, the appropriate ACTION statement(s) of those Limiting Conditions for Operation pertaining to the valve(s) or system in which it is installed shall be applicable.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Section A and B of Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of testing pursuant to Specification 4.0.5. Valves secured** in their actuated position are considered OPERABLE pursuant to this specification.

4.6.3.2 Each isolation valve specified in Section A and B of Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by verifying that on a ESFAS test signal, each isolation valve actuates to its isolation position.

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* Any flow path from the atmosphere or a piping system inside of containment to the atmosphere or a piping system outside of containment. Each flow path is considered as a separate "penetration".

** Locked, sealed or otherwise prevented from unintentional operation.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.3 The isolation time of each power operated or automatic valve in Section A and B of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 The manual isolation valves specified in Section C of Table 3.6-1 shall be demonstrated OPERABLE as required by 10 CFR 50, Appendix J and at least once per 31 days by verifying each valve# is secured** closed or blind flanged. Check valves specified in Section C of Table 3.6-1 shall be demonstrated OPERABLE pursuant to 10 CFR 50, Appendix J.

4.6.3.5 The isolation valves specified in Section D of Table 3.6-1 shall be demonstrated OPERABLE as required by Specification 4.0.5 and surveillance requirements associated with those Limiting Conditions for Operation pertaining to each valve or system in which it is installed. Valves secured** in their actuated position are considered OPERABLE pursuant to this specification.

#
Except valves and blind flanges, which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These valves and blind flanges shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

**
Locked, sealed or otherwise prevented from unintentional operation.

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
A.	CONTAINMENT ISOLATION (CIAS)		
1	HV-0510	Pressurizer steam space sample	40
1	HV-0511	Pressurizer steam space sample	40
2	TV-9267	Letdown line to letdown heat exchanger	40
2	HV-9205	Letdown line to letdown heat exchanger	40
4	HV-0508	Reactor coolant loops hot leg sample	40
4	HV-0509	Reactor coolant loops hot leg sample	40
4	HV-0517	Reactor coolant loops hot leg sample	40
6	HV-9334	Safety injection drain to RWST	40
7	HV-9217	Reactor coolant pump seal bleed off	40
7	HV-9218	Reactor coolant pump seal bleed off	40
11	HV-7911	Demineralized water to service station and sump pump	40
12	HV-0512	Pressurizer surge line sample	40
12	HV-0513	Pressurizer surge line sample	40
13	HV-5803	Containment sump pump discharge	40
13	HV-5804	Containment sump pump discharge	40
14	HV-5686	Fire protection	40
16C	HV-7805	Containment air radioactivity monitor inlet	1
16C	HV-7810	Containment air radioactivity monitor inlet	1
22	HV-5388	Instrument air supply line	40
23A	HV-5437	N_2 supply to quench tank, reactor coolant drain tank, and steam generators	40
26	HV-7512	Reactor coolant drain tank pump discharge	40
26	HV-7513	Reactor coolant drain tank pump discharge	40
27C	HV-7806	Containment air radioactivity monitor outlet	1
27C	HV-7811	Containment air radioactivity monitor outlet	1
30A	HV-7802	Containment air radioactivity monitor outlet	1
30A	HV-7803	Containment air radioactivity monitor outlet	1
30B	HV-7801	Containment air radioactivity monitor outlet	1
30B	HV-7800	Containment air radioactivity monitor outlet	1
30B	HV-7816	Containment air radioactivity monitor outlet	1
30C	HV-0516	Quench tank and drain tank gas sample	40
30C	HV-0514	Quench tank and drain tank gas sample	40
30C	HV-0515	Quench tank and drain tank gas sample	40
42	HV-6211	Component cooling water inlet	40
42	HV-6223	Component cooling water inlet	40
43	HV-6236	Component cooling water outlet	40
43	HV-6216	Component cooling water outlet	40
45	HV-9900	Containment normal A/C chilled water inlet	40
45	HV-9920	Containment normal A/C chilled water inlet	40
46	HV-9971	Containment normal A/C chilled water outlet	40
46	HV-9921	Containment normal A/C chilled water outlet	40
47	HV-7258	Containment waste gas vent header	40
47	HV-7259	Containment waste gas vent header	40
77	HV-5434	Nitrogen supply to safety injection tanks	40

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
B.	CONTAINMENT PURGE (CPIS)		
18	HV-9949**	Containment purge inlet (normal)	12
18	HV-9948**	Containment purge inlet (normal)	12
18	HV-9821	Containment mini-purge inlet	5
18	HV-9823	Containment mini-purge inlet	5
19	HV-9950**	Containment purge outlet (normal)	12
19	HV-9951**	Containment purge outlet (normal)	12
19	HV-9824	Containment mini-purge outlet	5
19	HV-9825	Containment mini-purge outlet	5

** Power to the valve removed in accordance with Specification 3.6.1.7.

C.	MANUAL*		
6	2"-099-C-334	Safety injection drain to RWST	NA
9	HV-9337	Shutdown cooling to LPSI pumps	NA
9	HV-9377	Shutdown cooling to LPSI pumps	NA
9	HV-9336	Shutdown cooling to LPSI pumps	NA
9	HV-9379	Shutdown cooling to LPSI pumps	NA
9	PSV-9349	Shutdown cooling relief	NA
10B	3/4"-038-C-396	Integrated leak rate test pressure sensor	NA
10B	3/4"-039-C-396	Integrated leak rate test pressure sensor	NA
11	3"-236-C-675	Demineralized water to service stations and sump pump	NA
14	4"-061-C-681	Fire protection	NA
16A	HV-0500	Post LOCA hydrogen monitor	NA
16A	HV-0501	Post LOCA hydrogen monitor	NA
16B	HV-0502	Post LOCA hydrogen monitor	NA
16B	HV-0503	Post LOCA hydrogen monitor	NA
20	2"-321-C-376	Quench tank makeup	NA
20	2"-573-C-611	Quench tank makeup	NA
21	2"-055-C-387	Service air supply line	NA
21	2"-017-C-627	Service air supply line	NA
22	1-1/2"-016-C-617	Instrument air supply line	NA
23A	3/4"-002-C-611	LP N_2 to containment	NA
25	10"-100-C-212	Refueling canal fill and drain	NA
25	10"-101-C-212	Refueling canal fill and drain	NA
31	HV-9946	Containment hydrogen purge inlet	NA
31	HCV-9945	Containment hydrogen purge inlet	NA

* Manual valves may be opened on an intermittent basis under administrative control.

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. MANUAL* (Cont.)			
68	2"-130-C-334	Charging line to auxiliary spray	NA
68	2"-129-A-554	Charging line to auxiliary spray	NA
70	2"-037-C-387	Auxiliary steam inlet to utility stations	NA
70	2"-038-C-387	Auxiliary steam inlet to utility stations	NA
74	HV-9917	Containment hydrogen purge outlet	NA
74	HCV-9918	Containment hydrogen purge outlet	NA
77	2"-108-C-627	Nitrogen supply to safety injection tanks	NA

* Manual valves may be opened on an intermittent basis under administrative control.

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
D. OTHER***			
3	3"-018-A-551	High pressure safety injection	NA
3	HV-9323	High pressure safety injection	NA
3	HV-9324	High pressure safety injection	NA
5	3"-019-A-551	High pressure safety injection	NA
5	HV-9326	High pressure safety injection	NA
5	HV-9327	High pressure safety injection	NA
8	2"-122-C-554	Charging line to regenerative heat exchanger	NA
8	HV-9200	Charging line to regenerative heat exchanger	NA
10A	HV-0352A	Containment pressure detectors	NA
27A	HV-0352D	Containment pressure detectors	NA
39	3"-020-A-551	High pressure safety injection	NA
39	HV-9329	High pressure safety injection	NA
39	HV-9330	High pressure safety injection	NA
40A	HV-0352B	Containment pressure detectors	NA
41	3"-021-A-551	High pressure safety injection	NA
41	HV-9332	High pressure safety injection	NA
41	HV-9333	High pressure safety injection	NA
48	8"-072-A-552	Low pressure safety injection	NA
48	HV-9322	Low pressure safety injection	NA
49	8"-073-A-552	Low pressure safety injection	NA
49	HV-9325	Low pressure safety injection	NA

*** Valves secured in the open (ESFAS actuated position) are considered OPERABLE pursuant to this specification.

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
D. OTHER*** (Cont.)			
50	8"-074-A-552	Low pressure safety injection	NA
50	HV-9328	Low pressure safety injection	NA
51	8"-075-A-552	Low pressure safety injection	NA
51	HV-9331	Low pressure safety injection	NA
52	8"-004-C-406	Containment spray inlet	NA
52	HV-9367	Containment spray inlet	NA
53	8"-006-C-406	Containment spray inlet	NA
53	HV-9368	Containment spray inlet	NA
54	HV-9304	Containment emergency sump recirculation	NA
54	HV-9302	Containment emergency sump recirculation	NA
55	HV-9305	Containment emergency sump recirculation	NA
55	HV-9303	Containment emergency sump recirculation	NA
56	HV-6366	Containment emergency A/C cooling water inlet	NA
57	HV-6372	Containment emergency A/C cooling water inlet	NA
58	HV-6368	Containment emergency A/C cooling water inlet	NA
59	HV-6370	Containment emergency A/C cooling water inlet	NA
60	HV-6369	Containment emergency A/C coling water outlet	NA
61	HV-6371	Containment emergency A/C cooling water outlet	NA
62	HV-6367	Containment emergency A/C cooling water outlet	NA
63	HV-6373	Containment emergency A/C cooling water outlet	NA
67	3"-157-A-550	Hot leg injection	NA
67	HV-9434	Hot leg injection	NA
71	3"-158-A-550	Hot leg injection	NA
71	HV-9420	Hot leg injection	NA
73A	HV-0352C	Containment pressure detectors	NA

***
Valves secured in the open (ESFAS actuated position) are considered OPERABLE pursuant to this specification.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. Such valves are listed in Sections A and B of Table 3.6-1 and Surveillance requirements to verify OPERABILITY of these valves are explicitly stated in 4.6.3.1 thru 4.6.3.3. Check valves located inside containment are considered OPERABLE provided their leak rate is within limits when tested pursuant to 10 CFR 50 Appendix J.

Section C of Table 3.6-1 contains a listing of manual and self actuated (check) valves that are normally closed and assumed to be closed under design basis accident conditions, but which may be opened intermittently for service, maintenance or test during normal operation provided adequate administrative controls are implemented to ensure operator action is taken to close such valves in the event of an accident.

All valves in Section A, B or C are considered OPERABLE for containment isolation purpose if they are indeed locked, sealed or otherwise secured in the closed position and leakage through the affected flow path is shown to be within limits when tested pursuant to 10 CFR 50 Appendix J.

Section D of Table 3.6-1 contains a listing of valves assumed to be open or to open automatically on an ESFAS signal to prevent or mitigate the consequences of the design basis accident. Surveillance requirement 4.6.1.1.a is not applicable to such valves. The OPERABILITY of such valves is determined by ESFAS response time testing of specification 3/4.3.2. Valves in Section D that are locked, sealed or otherwise secured in their open position will permit performance of their safety function and are, therefore, considered OPERABLE pursuant to this specification.

Attachment "C"

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

4.6.3.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a CIAS or SIAS test signal, each isolation valve actuates to its isolation position.

SURVEILLANCE REQUIREMENTS (Continued)

b. Verifying that on a Containment Radiation-High test signal, all containment purge valves actuate to their isolation position.

4.6.3.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
A.	CONTAINMENT ISOLATION (CIAS)		
1	HV-0510	Pressurizer steam space sample	40
1	HV-0511	Pressurizer steam space sample	40
2	TV-9267	Letdown line to letdown heat exchanger	40
2	HV-9205	Letdown line to letdown heat exchanger	40
4	HV-0508	Reactor coolant loops hot leg sample	40
4	HV-0509	Reactor coolant loops hot leg sample	40
4	HV-0517	Reactor coolant loops hot leg sample	40
6	HV-9334	Safety injection drain to RWST	40
7	HV-9217	Reactor coolant pump seal bleed off	40
7	HV-9218	Reactor coolant pump seal bleed off	40
11	HV-7911	Demineralized water to service station and sump pump	40
12	HV-0512	Pressurizer surge line sample	40
12	HV-0513	Pressurizer surge line sample	40
13	HV-5803	Containment sump pump discharge	40
13	HV-5804	Containment sump pump discharge	40
14	HV-5686	Fire protection	40
16C	HV-7805	Containment air radioactivity monitor inlet	1
16C	HV-7810	Containment air radioactivity monitor inlet	1
18	HV-9821	Containment minipurge inlet	5
18	HV-9823	Containment minipurge inlet	5
19	HV-9824	Containment minipurge outlet	5
19	HV-9825	Containment minipurge outlet	5
22	HV-5388	Instrument air supply line	40
23A	HV-5437	N_2 supply to quench tank, reactor coolant drain tank, and steam generators	40
26	HV-7512	Reactor coolant drain tank pump discharge	40
26	HV-7513	Reactor coolant drain tank pump discharge	40
27C	HV-7806	Containment air radioactivity monitor outlet	1
27C	HV-7811	Containment air radioactivity monitor outlet	1

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TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
28	HV-4052#	Steam generator feedwater	10
29	HV-4048#	Steam generator feedwater	10
30A	HV-7802	Containment air radioactivity monitor inlet	1
30A	HV-7803	Containment air radioactivity monitor inlet	1
30B	HV-7801	Containment air radioactivity monitor outlet	1
30B	HV-7800	Containment air radioactivity monitor outlet	1
30B	HV-7816	Containment air radioactivity monitor outlet	1
30C	HV-0516	Quench tank and drain tank gas sample	40
30C	HV-0514	Quench tank and drain tank gas sample	40
30C	HV-0515	Quench tank and drain tank gas sample	40
32	HV-8204#	Mainsteam isolation	5
33	HV-8205#	Mainsteam isolation	5
42	HV-6211	Component cooling water inlet	40
43	HV-6216	Component cooling water outlet	40
45	HV-9900	Containment normal A/C chilled water inlet	40
45	HV-9920	Containment normal A/C chilled water inlet	40
46	HV-9971	Containment normal A/C chilled water inlet	40
46	HV-9921	Containment normal A/C chilled water outlet	40
47	HV-7258	Containment waste gas vent header	40
47	HV-7259	Containment waste gas vent header	40
77	HV-5434	Nitrogen supply to safety injection tanks	40

B. CONTAINMENT PURGE (CPIS)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
18	HV-9949**	Containment purge inlet (normal)	12
18	HV-9948**	Containment purge inlet (normal)	12
18	HV-9821	Containment mini-purge inlet	5
18	HV-9823	Containment mini-purge inlet	5
19	HV-9950**	Containment purge outlet (normal)	12
19	HV-9951**	Containment purge outlet (normal)	12
19	HV-9824	Containment mini-purge outlet	5
19	HV-9825	Containment mini-purge outlet	5

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. MANUAL			
6	2"-099-C-334*	Safety injection drain to RWST	NA
8	HV-9200	Charging line to regenerative heat exchanger	NA
9	HV-9337#@	Shutdown cooling to LPSI pumps	NA
9	HV-9377#@	Shutdown cooling to LPSI pumps	NA
9	HV-9336#@	Shutdown cooling to LPSI pumps	NA
9	HV-9379#@	Shutdown cooling to LPSI pumps	NA
10A	HV-0352A#	Containment pressure detectors	NA
10C	3/4"-038-C-396	Integrated leak rate test pressure sensor	NA
10C	3/4"-039-C-396	Integrated leak rate test pressure sensor	NA
16A	HV-0500*	Post LOCA hydrogen monitor	NA
16A	HV-0501*	Post LOCA hydrogen monitor	NA
16B	HV-0502*	Post LOCA hydrogen monitor	NA
16B	HV-0503*	Post LOCA hydrogen monitor	NA
20	2"-321-C-376*	Quench tank makeup	NA
21	2"-055-C-387	Service air supply line	NA
25	10"-100-C-212	Refueling canal fill and drain	NA
25	10"-101-C-212	Refueling canal fill and drain	NA
27A	HV-0352D#	Containment pressure detectors	NA
31	HV-9946	Containment hydrogen purge inlet	NA
31	HCV-9945	Containment hydrogen purge inlet	NA
40A	HV-0352B#	Containment pressure detectors	NA
67	HV-9434	Hot leg injection	NA
68	2"-130-C-334	Charging line to auxiliary spray	NA
70	2"-037-C-387	Auxiliary steam inlet to utility stations	NA
70	2"-038-C-387	Auxiliary steam inlet to utility stations	NA
71	HV-9420	Hot leg injection	NA
73A	HV-0352C#	Containment pressure detectors	NA
74	HV-9917	Containment hydrogen purge outlet	NA
74	HCV-9918	Containment hydrogen purge outlet	NA

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TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
D. OTHER			
3	3"-018-A-551#	High pressure safety injection	NA
3	HV-9323#	High pressure safety injection	NA
3	HV-9324#	High pressure safety injection	NA
5	3"-019-A-551#	High pressure safety injection	NA
5	HV-9326#	High pressure safety injection	NA
5	HV-9327#	High pressure safety injection	NA
8	2"-122-C-554	Charging line to regenerative heat exchanger	NA
9	PSV-9349#	Shutdown cooling relief	NA
11	3"-236-C-675	Demineralized water to service stations and sump pump	NA
14	4"-061-C-681	Fire protection	NA
17	HV-4058#^	Steam generator secondary coolant sample	NA
20	2"-573-C-611	Quench tank makeup	NA
21	2"-017-C-627	Service air supply line	NA
22	1-1/2"-016-C-617	Instrument air supply line	NA
23A	3/4"-002-C-611	LP N_2 to containment	NA
32	HV-8421#	Mainsteam atmospheric dump	NA
32	PSV-8410#	Mainsteam relief	NA
32	PSV-8411#	Mainsteam relief	NA
32	PSV-8412#	Mainsteam relief	NA
32	PSV-8413#	Mainsteam relief	NA
32	PSV-8414#	Mainsteam relief	NA
32	PSV-8415#	Mainsteam relief	NA
32	PSV-8416#	Mainsteam relief	NA
32	PSV-8417#	Mainsteam relief	NA
32	PSV-8418#	Mainsteam relief	NA
32	HV-8249B#	Mainsteam trap isolation	NA
32	HV-8202#	Mainsteam isolation bypass	NA
32	HV-8200#	Mainsteam to auxiliary feedwater turbine	NA
33	HV-8419#	Mainsteam atmospheric dump	NA
33	PSV-8401#	Mainsteam relief	NA

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
33	PSV-8402#	Mainsteam relief	NA
33	PSV-8403#	Mainsteam relief	NA
33	PSV-8404#	Mainsteam relief	NA
33	PSV-8405#	Mainsteam relief	NA
33	PSV-8406#	Mainsteam relief	NA
33	PSV-8407#	Mainsteam relief	NA
33	PSV-8408#	Mainsteam relief	NA
33	PSV-8409#	Mainsteam relief	NA
33	HV-8248B#	Mainsteam trap isolation	NA
33	HV-8203#	Mainsteam isolation bypass	NA
33	HV-8201#	Mainsteam to auxiliary feedwater turbine	NA
36	HV-4054#*	Steam generator blowdown	NA
37	HV-4053#*	Steam generator blowdown	NA
39	3"-020-A-551#	High pressure safety injection	NA
39	HV-9329#	High pressure safety injection	NA
39	HV-9330#	High pressure safety injection	NA
41	3"-021-A-551#	High pressure safety injection	NA
41	HV-9332#	High pressure safety injection	NA
41	HV-9333#	High pressure safety injection	NA
42	HV-6223	Component cooling water inlet	NA
43	HV-6236	Component cooling water inlet	NA
44	HV-4057#*	Steam generator secondary coolant sample	NA
48	8"-072-A-552#@	Low pressure safety injection	NA
48	HV-9322#@	Low pressure safety injection	NA
49	8"-073-A-552#@	Low pressure safety injection	NA
49	HV-9325#@	Low pressure safety injection	NA
50	8"-074-A-552#@	Low pressure safety injection	NA
50	HV-9328#@	Low pressure safety injection	NA
51	8"-075-A-552#@	Low pressure safety injection	NA
51	HV-9331#@	Low pressure safety injection	NA
52	8"-004-C-406	Containment spray inlet	NA
52	HV-9367	Containment spray inlet	NA

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
53	8"-006-C-406	Containment spray inlet	NA
53	HV-9368	Containment spray inlet	NA
54	HV-9304#	Containment emergency sump recirculation	NA
54	HV-9302#	Containment emergency sump recirculation	NA
55	HV-9305#	Containment emergency sump recirculation	NA
55	HV-9303#	Containment emergency sump recirculation	NA
56	HV-6366	Containment emergency A/C cooling water inlet	NA
57	HV-6372	Containment emergency A/C cooling water inlet	NA
58	HV-6368	Containment emergency A/C cooling water inlet	NA
59	HV-6370	Containment emergency A/C cooling water inlet	NA
60	HV-6369	Containment emergency A/C cooling water inlet	NA
61	HV-6371	Containment emergency A/C cooling water inlet	NA
62	HV-6367	Containment emergency A/C cooling water inlet	NA
63	HV-6373	Containment emergency A/C cooling water inlet	NA
67	3"-157-A-551	Hot leg injection	NA
68	2"-129-A-554	Charging line to auxiliary spray	NA
71	3"-158-A-551	Hot leg injection	NA
75	HV-4715#	Steam generator auxiliary feedwater	NA
75	HV-4731#	Steam generator auxiliary feedwater	NA
77	2"-108-C-627	Nitrogen supply to safety injection tanks	NA
78	HV-4714#	Steam generator auxiliary feedwater	NA
78	HV-4730#	Steam generator auxiliary feedwater	NA

*May be opened on an intermittent basis under administrative control.

**Power to the valve removed in accordance with Specification 3.6.1.7.

#Not subject to Type C leakage tests.

@Shutdown cooling valves may be opened in MODE 4.

3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the iodine removal system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure that the solution recirculated within containment after a LOCA has a pH value between 8.0 and 10.0 at the end of the NaOH injection period. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

The 5-year Surveillance testing is intended to verify that no crystallization of the NaOH or other obstruction has occurred in the piping from the spray additive tank to the suction of the containment spray pumps.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

The containment cooling system and the containment spray system are redundant to each other in providing post-accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the containment cooling system have been appropriately adjusted. However, the allowable out-of-service time requirements for the containment spray system have been maintained consistent with that assigned other inoperable ESF equipment since the containment spray system also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

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Attachment "D"

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

1. With one or more of the isolation valve(s) specified in Section A, B and C of Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration* that is open and either:

    a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

    b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured** in the isolation position, or one closed manual valve or blind flange, or

    c. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. With one or more of the valves specified in Section D of Table 3.6-1 inoperable, the appropriate ACTION statement(s) of those Limiting Conditions for Operation pertaining to the valve(s) or system in which it is installed shall be applicable.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Section A and B of Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of testing pursuant to Specification 4.0.5. Valves secured** in their actuated position are considered OPERABLE pursuant to this specification.

4.6.3.2 Each isolation valve specified in Section A and B of Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by verifying that on a ESFAS test signal, each isolation valve actuates to its isolation position.

---

* Any flow path from the atmosphere or a piping system inside of containment to the atmosphere or a piping system outside of containment. Each flow path is considered as a separate "penetration".

** Locked, sealed or otherwise prevented from unintentional operation.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.3 The isolation time of each power operated or automatic valve in Section A and B of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 The manual isolation valves specified in Section C of Table 3.6-1 shall be demonstrated OPERABLE as required by 10 CFR 50, Appendix J and at least once per 31 days by verifying each valve# is secured** closed or blind flanged. Check valves specified in Section C of Table 3.6-1 shall be demonstrated OPERABLE pursuant to 10 CFR 50, Appendix J.

4.6.3.5 The isolation valves specified in Section D of Table 3.6-1 shall be demonstrated OPERABLE as required by Specification 4.0.5 and surveillance requirements associated with those Limiting Conditions for Operation pertaining to each valve or system in which it is installed. Valves secured** in their actuated position are considered OPERABLE pursuant to this specification.

#
Except valves and blind flanges, which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These valves and blind flanges shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

**
Locked, sealed or otherwise prevented from unintentional operation.

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
A. CONTAINMENT ISOLATION (CIAS)			
1	HV-0510	Pressurizer steam space sample	40
1	HV-0511	Pressurizer steam space sample	40
2	TV-9267	Letdown line to letdown heat exchanger	40
2	HV-9205	Letdown line to letdown heat exchanger	40
4	HV-0508	Reactor coolant loops hot leg sample	40
4	HV-0509	Reactor coolant loops hot leg sample	40
4	HV-0517	Reactor coolant loops hot leg sample	40
6	HV-9334	Safety injection drain to RWST	40
7	HV-9217	Reactor coolant pump seal bleed off	40
7	HV-9218	Reactor coolant pump seal bleed off	40
11	HV-7911	Demineralized water to service station and sump pump	40
12	HV-0512	Pressurizer surge line sample	40
12	HV-0513	Pressurizer surge line sample	40
13	HV-5803	Containment sump pump discharge	40
13	HV-5804	Containment sump pump discharge	40
14	HV-5686	Fire protection	40
16C	HV-7805	Containment air radioactivity monitor inlet	1
16C	HV-7810	Containment air radioactivity monitor inlet	1
22	HV-5388	Instrument air supply line	40
23A	HV-5437	N_2 supply to quench tank, reactor coolant drain tank, and steam generators	40
26	HV-7512	Reactor coolant drain tank pump discharge	40
26	HV-7513	Reactor coolant drain tank pump discharge	40
27C	HV-7806	Containment air radioactivity monitor outlet	1
27C	HV-7811	Containment air radioactivity monitor outlet	1
30A	HV-7802	Containment air radioactivity monitor outlet	1
30A	HV-7803	Containment air radioactivity monitor outlet	1
30B	HV-7801	Containment air radioactivity monitor outlet	1
30B	HV-7800	Containment air radioactivity monitor outlet	1
30B	HV-7816	Containment air radioactivity monitor outlet	1
30C	HV-0516	Quench tank and drain tank gas sample	40
30C	HV-0514	Quench tank and drain tank gas sample	40
30C	HV-0515	Quench tank and drain tank gas sample	40
42	HV-6211	Component cooling water inlet	40
42	HV-6223	Component cooling water inlet	40
43	HV-6236	Component cooling water outlet	40
43	HV-6216	Component cooling water outlet	40
45	HV-9900	Containment normal A/C chilled water inlet	40
45	HV-9920	Containment normal A/C chilled water inlet	40
46	HV-9971	Containment normal A/C chilled water outlet	40
46	HV-9921	Containment normal A/C chilled water outlet	40
47	HV-7258	Containment waste gas vent header	40
47	HV-7259	Containment waste gas vent header	40
77	HV-5434	Nitrogen supply to safety injection tanks	40

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
B. CONTAINMENT PURGE (CPIS)			
18	HV-9949**	Containment purge inlet (normal)	12
18	HV-9948**	Containment purge inlet (normal)	12
18	HV-9821	Containment mini-purge inlet	5
18	HV-9823	Containment mini-purge inlet	5
19	HV-9950**	Containment purge outlet (normal)	12
19	HV-9951**	Containment purge outlet (normal)	12
19	HV-9824	Containment mini-purge outlet	5
19	HV-9825	Containment mini-purge outlet	5

** Power to the valve removed in accordance with Specification 3.6.1.7.

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. MANUAL*			
6	2"-099-C-334	Safety injection drain to RWST	NA
9	HV-9337	Shutdown cooling to LPSI pumps	NA
9	HV-9377	Shutdown cooling to LPSI pumps	NA
9	HV-9336	Shutdown cooling to LPSI pumps	NA
9	HV-9379	Shutdown cooling to LPSI pumps	NA
9	PSV-9349	Shutdown cooling relief	NA
10C	3/4"-038-C-396	Integrated leak rate test pressure sensor	NA
10C	3/4"-039-C-396	Integrated leak rate test pressure sensor	NA
11	3"-236-C-675	Demineralized water to service stations and sump pump	NA
14	4"-061-C-681	Fire protection	NA
16A	HV-0500	Post LOCA hydrogen monitor	NA
16A	HV-0501	Post LOCA hydrogen monitor	NA
16B	HV-0502	Post LOCA hydrogen monitor	NA
16B	HV-0503	Post LOCA hydrogen monitor	NA
20	2"-321-C-376	Quench tank makeup	NA
20	2"-573-C-611	Quench tank makeup	NA
21	2"-055-C-387	Service air supply line	NA
21	2"-017-C-627	Service air supply line	NA
22	1-1/2"-016-C-617	Instrument air supply line	NA
23A	3/4"-002-C-611	LP N_2 to containment	NA
25	10"-100-C-212	Refueling canal fill and drain	NA
25	10"-101-C-212	Refueling canal fill and drain	NA
31	HV-9946	Containment hydrogen purge inlet	NA
31	HCV-9945	Containment hydrogen purge inlet	NA

* Manual valves may be opened on an intermittent basis under administrative control.

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. MANUAL* (Cont.)			
68	2"-130-C-334	Charging line to auxiliary spray	NA
68	2"-129-A-554	Charging line to auxiliary spray	NA
70	2"-037-C-387	Auxiliary steam inlet to utility stations	NA
70	2"-038-C-387	Auxiliary steam inlet to utility stations	NA
74	HV-9917	Containment hydrogen purge outlet	NA
74	HCV-9918	Containment hydrogen purge outlet	NA
77	2"-108-C-627	Nitrogen supply to safety injection tanks	NA

* Manual valves may be opened on an intermittent basis under administrative control.

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
D. OTHER***			
3	3"-018-A-551	High pressure safety injection	NA
3	HV-9323	High pressure safety injection	NA
3	HV-9324	High pressure safety injection	NA
5	3"-019-A-551	High pressure safety injection	NA
5	HV-9326	High pressure safety injection	NA
5	HV-9327	High pressure safety injection	NA
8	2"-122-C-554	Charging line to regenerative heat exchanger	NA
8	HV-9200	Charging line to regenerative heat exchanger	NA
10A	HV-0352A	Containment pressure detectors	NA
27A	HV-0352D	Containment pressure detectors	NA
39	3"-020-A-551	High pressure safety injection	NA
39	HV-9329	High pressure safety injection	NA
39	HV-9330	High pressure safety injection	NA
40A	HV-0352B	Containment pressure detectors	NA
41	3"-021-A-551	High pressure safety injection	NA
41	HV-9332	High pressure safety injection	NA
41	HV-9333	High pressure safety injection	NA
48	8"-072-A-552	Low pressure safety injection	NA
48	HV-9322	Low pressure safety injection	NA
49	8"-073-A-552	Low pressure safety injection	NA
49	HV-9325	Low pressure safety injection	NA

*** Valves secured in the open (ESFAS actuated position) are considered OPERABLE pursuant to this specification.

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
D. OTHER*** (Cont.)			
50	8"-074-A-552	Low pressure safety injection	NA
50	HV-9328	Low pressure safety injection	NA
51	8"-075-A-552	Low pressure safety injection	NA
51	HV-9331	Low pressure safety injection	NA
52	8"-004-C-406	Containment spray inlet	NA
52	HV-9367	Containment spray inlet	NA
53	8"-006-C-406	Containment spray inlet	NA
53	HV-9368	Containment spray inlet	NA
54	HV-9304	Containment emergency sump recirculation	NA
54	HV-9302	Containment emergency sump recirculation	NA
55	HV-9305	Containment emergency sump recirculation	NA
55	HV-9303	Containment emergency sump recirculation	NA
56	HV-6366	Containment emergency A/C cooling water inlet	NA
57	HV-6372	Containment emergency A/C cooling water inlet	NA
58	HV-6368	Containment emergency A/C cooling water inlet	NA
59	HV-6370	Containment emergency A/C cooling water inlet	NA
60	HV-6369	Containment emergency A/C coling water outlet	NA
61	HV-6371	Containment emergency A/C cooling water outlet	NA
62	HV-6367	Containment emergency A/C cooling water outlet	NA
63	HV-6373	Containment emergency A/C cooling water outlet	NA
67	3"-157-A-550	Hot leg injection	NA
67	HV-9434	Hot leg injection	NA
71	3"-158-A-550	Hot leg injection	NA
71	HV-9420	Hot leg injection	NA
73A	HV-0352C	Containment pressure detectors	NA

***
Valves secured in the open (ESFAS actuated position) are considered OPERABLE pursuant to this specification.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA. Such valves are listed in Sections A and B of Table 3.6-1 and Surveillance requirements to verify OPERABILITY of these valves are explicitly stated in 4.6.3.1 thru 4.6.3.3. Check valves located inside containment are considered OPERABLE provided their leak rate is within limits when tested pursuant to 10 CFR 50 Appendix J.

Section C of Table 3.6-1 contains a listing of manual and self actuated (check) valves that are normally closed and assumed to be closed under design basis accident conditions, but which may be opened intermittently for service, maintenance or test during normal operation provided adequate administrative controls are implemented to ensure operator action is taken to close such valves in the event of an accident.

All valves in Section A, B or C are considered OPERABLE for containment isolation purpose if they are indeed locked, sealed or otherwise secured in the closed position and leakage through the affected flow path is shown to be within limits when tested pursuant to 10 CFR 50 Appendix J.

Section D of Table 3.6-1 contains a listing of valves assumed to be open or to open automatically on an ESFAS signal to prevent or mitigate the consequences of the design basis accident. Surveillance requirement 4.6.1.1.a is not applicable to such valves. The OPERABILITY of such valves is determined by ESFAS response time testing of specification 3/4.3.2. Valves in Section D that are locked, sealed or otherwise secured in their open position will permit performance of their safety function and are, therefore, considered OPERABLE pursuant to this specification.

DESCRIPTION OF PROPOSED CHANGES NPF-10-108 AND NPF-15-108 AND SAFETY ANALYSIS

This is a request to revise Technical Specification Table 3.8-1, "Containment Penetration Conductor Overcurrent Protective Devices."

Existing Specifications:

Units 2 and 3: See Attachment "A"

Proposed Specifications:

Units 2 and 3: See Attachment "B"

Discussion:

This change is strictly an editorial change to correct equipment designations incorrectly shown in Table 3.8-1,"Containment Penetration Conductor Overcurrent Protective Devices," for both Unit 2 and Unit 3. The overcurrent protective devices and the equipment they served were correctly designated but some of the equipment designation numbers for the served equipment were incorrect. The correct designations are shown in Attachment "B"

Southern California Edison (SCE) letter dated January 14, 1983 describes equipment designations noted in the subject table. Unit 2 Table 3.8-1 currently shows Containment Recirculation Unit designated E-333 which is in error. Unit 3 Table 3.8-1 currently shows Standby Containment Normal Cooling Fan designated E-333, Containment Normal Cooling Fan designated E-334, and Containment Recirculation Unit designated E-333 all of which are in error.

The proposed changes revises Table 3.8-1 to read the proper equipment designations. Unit 2 Table 3.8-1 should read Containment Recirculation Unit A-353. Unit 3 Table 3.8-1 should read Standby Containment Normal Cooling Fan E-393, Containment Normal Cooling Fan E-394, and Containment Recirculation Unit A-353.

Safety Analysis:

The proposed change discussed above shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1) Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Safety Analysis (continued):

Response: No

The changes proposed to Table 3.8-1 are to correct equipment designations and do not affect the probability or consequences of accidents.

2) Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The changes proposed to Table 3.8-1 are to correct equipment designations and do not create the possibility of a new or different kind of accident.

3) Will Operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

Response: No

The changes proposed to Table 3.8-1 are to correct equipment designations and do not involve a reduction in a margin of safety.

The proposed revision of Table 3.8-1 is similar to example (i) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983, in that it is essentially administrative in nature.

Safety and Significant Hazards Determination:

Based on the Safety Analysis, it is concluded that: 1) The proposed change does not involve a significant hazards consideration as defined by 10CFR50.92; 2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and, 3) this action will not result in a condition which significantly alters the impact of the Station on the environment as described in the NRC Final Environmental Statement.

ATTACHMENT B

(Proposed Specification)

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2B0106	2BLP0101	Containment Normal Cooling Fan E-397
2B0107	2BLP0102	CEDM Cooling Supply Fan E-403B
2B0109	2BLP0103	CEDM Cooling Supply Fan E-403A
2B0111	2BLP0104	Standby Containment Normal Cooling Fan E-393
2B0209	2BLP0201	Containment Normal Cooling Fan E-394
2B0406	2BLP0301	Hydrogen Recombiner E-145 Power Panel L-180
2B0409	2BLP0302	Upper Dome Air Circulator A-071
2B0410	2BLP0303	Containment Emergency Fan E-399
2B0411	2BLP0304	Containment Emergency Fan E-401
2B0419	2BLP0305	Standby Upper Dome Air Circulator A-074
2B0606	2BLP0401	Hydrogen Recombiner E-146 Power Panel L-181
2B0609	2BLP0402	Upper Dome Air Circulator A-072
2B0610	2BLP0403	Containment Emergency Fan E-400
2B0611	2BLP0404	Containment Emergency Fan E-402
2B0619	2BLP0405	Standby Upper Dome Air Circulator A-073
2B0809	2BLP0501	Containment Normal Cooling Fan E-396
2B0811	2BLP0601	Containment Normal Cooling Fan E-398
2B0903	2BLP0701	Containment Recirculation Unit A-353
2B0906	2BLP0702	Polar Crane (Containment) R001 (C)
2B0907	2BLP0703	Standby Control Element Drive Mechanism Cooling Supply Fan E-404A
2B0909	2BLP0704	Standby CEDM Cooling Supply Fan E-404B
2B0911	2BLP0705	Containment Recirculating Unit Heater E-560
2BA02	2BLP0812	CCW from RCP P-001 Seal Heat Exchanger TV-9144
2BA03	2BLP0813	CCW from RCP P-003 Seal Heat Exchanger TV-9154
2BA04 (2BA04-A)	2BLP0801	CEDM Cooling Supply Fan E-403A (Enclosure Heater)

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
3B0106	3BLP0101	Containment Normal Cooling Fan E-397
3B0107	3BLP0102	CEDM Cooling Supply Fan E-403B
3B0109	3BLP0103	CEDM Cooling Supply Fan E-403A
3B0111	3BLP0104	Standby Containment Normal Cooling Fan E-393
3B0209	3BLP0201	Containment Normal Cooling Fan E-394
3B0406	3BLP0301	Hydrogen Recombiner E-145 Power Panel L-180
3B0409	3BLP0302	Upper Dome Air Circulator A-701
3B0410	3BLP0303	Containment Emergency Fan E-399
3B0411	3BLP0304	Containment Emergency Fan E-401
3B0419	3BLP0305	Standby Upper Dome Air Circulator A-074
3B0606	3BLP0401	Hydrogen Recombiner E-146 Power Panel L-181
3B0609	3BLP0402	Upper Dome Air Circulator A-072
3B0610	3BLP0403	Containment Emergency Fan E-400
3B0611	3BLP0404	Containment Emergency Fan E-402
3B0619	3BLP0405	Standby Upper Dome Air Circulator A-073
3B0809	3BLP0501	Containment Normal Cooling Fan E-396
3B0811	3BLP0601	Containment Normal Cooling Fan E-398
3B0903	3BLP0701	Containment Recirculation Unit A-353
3B0906	3BLP0702	Polar Crane (Containment) R001 (C)
3B0907	3BLP0703	Standby Control Element Drive Mechanism Cooling Supply Fan E-404A
3B0909	3BLP0704	Standby CEDM Cooling Supply Fan E-404B
3B0911	3BLP0705	Containment Recirculating Unit Heater E-568
3BA02	3BLP0812	CCW from RCP P-001 Seal Heat Exchanger TV-9144
3BA03	3BLP0813	CCW from RCP P-003 Seal Heat Exchanger TV-9154
3BA04 (3BA04-A)	3BLP0801	CEDM Cooling Supply Fan E-403A (Enclosure Heater)

ATTACHMENT A

(Existing Specification)

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2B0106	2BLP0101	Containment Normal Cooling Fan E-397
2B0107	2BLP0102	CEDM Cooling Supply Fan E-403B
2B0109	2BLP0103	CEDM Cooling Supply Fan E-403A
2B0111	2BLP0104	Standby Containment Normal Cooling Fan E-393
2B0209	2BLP0201	Containment Normal Cooling Fan E-394
2B0406	2BLP0301	Hydrogen Recombiner E-145 Power Panel L-180
2B0409	2BLP0302	Upper Dome Air Circulator A-071
2B0410	2BLP0303	Containment Emergency Fan E-399
2B0411	2BLP0304	Containment Emergency Fan E-401
2B0419	2BLP0305	Standby Upper Dome Air Circulator A-074
2B0606	2BLP0401	Hydrogen Recombiner E-146 Power Panel L-181
2B0609	2BLP0402	Upper Dome Air Circulator A-072
2B0610	2BLP0403	Containment Emergency Fan E-400
2B0611	2BLP0404	Containment Emergency Fan E-402
2B0619	2BLP0405	Standby Upper Dome Air Circulator A-073
2B0809	2BLP0501	Containment Normal Cooling Fan E-396
2B0811	2BLP0601	Containment Normal Cooling Fan E-398
2B0903	2BLP0701	Containment Recirculation Unit E-333
2B0906	2BLP0702	Polar Crane (Containment) R001 (C)
2B0907	2BLP0703	Standby Control Element Drive Mechanism Cooling Supply Fan E-404A
2B0909	2BLP0704	Standby CEDM Cooling Supply Fan E-404B
2B0911	2BLP0705	Containment Recirculating Unit Heater E-568
2BA02	2BLP0812	CCW from RCP P-001 Seal Heat Exchanger TV-9144
2BA03	2BLP0813	CCW from RCP P-003 Seal Heat Exchanger TV-9154
2BA04 (2BA04-A)	2BLP0801	CEDM Cooling Supply Fan E-403A (Enclosure Heater)

ISSUED TO A CONTROLLED LOCATION

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
3B0106	3BLP0101	Containment Normal Cooling Fan E-397
3B0107	3BLP0102	CEDM Cooling Supply Fan E-403B
3B0109	3BLP0103	CEDM Cooling Supply Fan E-403A
3B0111	3BLP0104	Standby Containment Normal Cooling Fan E-333
3B0209	3BLP0201	Containment Normal Cooling Fan E-334
3B0406	3BLP0301	Hydrogen Recombiner E-145 Power Panel L-180
3B0409	3BLP0302	Upper Dome Air Circulator A-701
3B0410	3BLP0303	Containment Emergency Fan E-399
3B0411	3BLP0304	Containment Emergency Fan E-401
3B0419	3BLP0305	Standby Upper Dome Air Circulator A-074
3B0606	3BLP0401	Hydrogen Recombiner E-146 Power Panel L-181
3B0609	3BLP0402	Upper Dome Air Circulator A-072
3B0610	3BLP0403	Containment Emergency Fan E-400
3B0611	3BLP0404	Containment Emergency Fan E-402
3B0619	3BLP0405	Standby Upper Dome Air Circulator A-073
3B0809	3BLP0501	Containment Normal Cooling Fan E-396
3B0811	3BLP0601	Containment Normal Cooling Fan E-398
3B0903	3BLP0701	Containment Recirculation Unit E-333
3B0906	3BLP0702	Polar Crane (Containment) R001 (C)
3B0907	3BLP0703	Standby Control Element Drive Mechanism Cooling Supply Fan E-404A
3B0909	3BLP0704	Standby CEDM Cooling Supply Fan E-404B
3B0911	3BLP0705	Containment Recirculating Unit Heater E-568
3BA02	3BLP0812	CCW from RCP P-001 Seal Heat Exchanger TV-9144
3BA03	3BLP0813	CCW from RCP P-003 Seal Heat Exchanger TV-9154
3BA04 (3BA04-A)	3BLP0801	CEDM Cooling Supply Fan E-403A (Enclosure Heater)

CONTROLLED ISSUED TO A

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Description of Proposed Changes NPF-10-111
and NPF-15-111 and Safety Analysis

This is a request to revise Section 3/4.11.2, Table 4.11-2 "RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM" of the Technical Specifications and to add a new Specification, 3.11.2.7 "RADIOACTIVE EFFLUENTS, AUXILIARY BOILER" for San Onofre Nuclear Generating Station, Units 2 and 3.

Existing Specification:

Units 2 and 3: See Attachment "A"

Proposed Specifications:

Units 2 and 3: See Attachment "B"

Description

An exemption is being sought to 10 CFR 20.305 by means of a Technical Specification change to allow disposal of radioactively contaminated Reactor Coolant Pump (RCP) motor oil, turbine building sump, and other waste oil by incineration.

Background

Reactor coolant pump motor oil is exposed to the containment atmosphere by means of breather lines and becomes slightly contaminated during the service life. It is also possible to introduce small amounts of contamination during handling. In addition to the RCP motor oil, small amounts of slightly contaminated lubricating oil are occasionally separated out of the water and other matter in plant sumps.

In the past, oil from these sources at San Onofre Nuclear Generating Station (SONGS) Unit 1 was mixed with adsorbent material in 55 gallon drums, and was shipped to Richland, Washington for radwaste burial. Such shipments are expensive and use the limited space available in burial sites.

Reactor coolant pump oil at SONGS 1 is drained and refilled during refueling outages which occur approximately every two years. Assuming the same frequency for SONGS 2/3, the projected volume of contaminated oil produced every two years will be:

Unit	Gallons Oil Per Pump	Number of Pumps	Volume (gallons)
1	225	3	675
2	150	4	600
3	150	4	600
			1875 Total

If only one 55 gallon drum of slightly contaminated oil from sumps were collected each year, the total volume of oil from both sources would not be expected to exceed approximately 2,000 gallons in any two year period.

Radiation Levels

Waste RCP motor oil produced at San Onofre Unit 1 has been found with contamination levels so low that the total activity in each drum was well within the limits of 10 CFR 20 Appendix C. Used oil is drained from the motors directly into 55 gallon drums. A sample is taken from each drum and all samples are isotopically analyzed. The last shipment of RCP motor oil occurred on August 13, 1982. That oil contained only cesium-137 and cobalt-60. However, cobalt-58 and cesium-134 are also likely to be contaminants and might be observed in future batches of waste oil from any of the three units at SONGS. Table 1 lists results of analyses of samples obtained from the August 13, 1982 shipment from San Onofre. In an effort to estimate the contamination levels expected in RCP oil from SONGS Units 2 and 3 (where the first oil changes have not yet occurred), data from one other utility was acquired and is included as Table 2. Contamination levels listed on Table 2 are, on the average, two orders of magnitude higher that those from SONGS-1, yet still must be considered very low level waste.

Proposed Method of Disposal

It is proposed that waste oil be burned in the SONGS 2/3 auxiliary boiler. The boiler is Combustion Engineering model number 85575, manufactured in 1976, with a capacity of 180,000 pounds of steam per hour. Number 2 fuel oil is fed to the boiler at a rate of 38 gallons per minute through supply line SA2421ML041. The waste RCP lubricating oil (Chevron GST68) would be fed into the same line upstream of strainers on the suction side of the fuel oil pumps (P084 and P085). Waste oil would be pumped directly out of the 55 gallon drum

at a fixed rate and would be fed into the supply line through either the existing vacuum pump connection for priming, or through a tee to be installed in the line just for this purpose. Waste oil would be mixed with fuel oil, would be burned in the boiler, and the exhaust would be diluted by the flow of air (42,966 cubic feet per minute) supplied by forced draft fan A409. Since the total activity in each drum, oil feed rate, and dilution factors would be known, the exact magnitude of each of these effluent releases could easily be calculated. The calculational methodology will be controlled and given in the ODCM.

Alternatives to Incineration

The following alternatives were considered during the formulation of this request:

1. Continue to package and ship waste oil containing exempt quantities of radionuclides to a low level radwaste burial site.

   a. Both adsorption and solidification of the waste oil before shipment for burial significantly increases the volume to be buried. For example, depending on the method used, solidifcation would increase the volume by a factor of 2-5. This is contrary to the industry's efforts to minimize solid waste volume and is not the best use for the limited burial site space.

   b. Packaging and shipping such low level waste to a burial site is expensive.

2. Accumulate and store waste oil onsite.

   a. Storage in drums would present a contamination control problem for Health Physics, a radwaste inventory burden for the Radwaste group, a potential fire hazard, and would occupy already limited space. This only prolongs the time before a final resolution to the problem is accepted.

   b. Construction of a storage tank would not be desirable because segregation of oil with differing amounts of activity would not be possible. The problem of disposal would eventually have to be faced when the tank was full, and, by that time, the burial cost would probably have increased.

Justification for Burning

The advantages of burning over the other alternatives are as follows:

1. Burning would eliminate the need for packaging and transporting the material to a burial site.

2. Burning would not use any of the limited burial space available, and would reduce the volume of solid radwaste produced.

3. The total radioactivity content of the oil is low. Effluent released to the environment would be insignificant because the oil would be fed to the boiler from each drum indiviudally, and would be diluted by fuel oil. The concentrations released would be very low since the total flow out the stack is approximately 42,966 cfm. If the proposed method is used, the concentration of radioactivity leaving the stack every second will be easy to calculate and document, per methods in the Offsite Dose Calculation Manual (ODCM), and will remain decades below the limits set forth in Column 1 of Table II in Appendix B. ODCM methodology will ensure that the addition of this pathway will not cause the total of all releases from the site to exceed the dose rate objectives of 10 CFR 50, Appendix I. Because both the concentrations and total radioactivity content of the oil are very low, the boiler exhaust clearly would not be a principle gaseous effluent discharge path. Continuous sampling and monitoring would, therefore, not be required

4. Burning would dispose of the waste and produce useful energy in the process.

Summary

It is concluded that burning exempt quantities of waste RCP and sump oil in the SONGS 2/3 auxiliary boiler would be the most judicious method of disposal. It is requested that a Technical Specification change and an exemption to 10 CFR 20.305, be approved to implement this disposal method utilizing the provision of 10 CFR 20.302 and IE Information Notice No. 83-05 "Obtaining approval for disposing of very low radioactive waste - 10 CFR Section 20.302".

Safety Analysis

The proposed change discussed above shall be deemed to constitute a significant hazards consideration if there is a positive finding in any of the following areas.

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

The release of any radioactive material contained in the oil does not constitute an accident evaluated in the FSAR. Compliance with 10 CFR 20 Appendix B, Table II Column I, and dose objectives of 10 CFR 50, Appendix I, insures that the health and safety of the public will not be endangered and that there will be no significant impact by the Station on the environment. Concentrations of any radioactivity leaving the Station will be calculated and documented per methods in the ODCM. The potential dose that could occur as a result of the incineration of contaminated oil has been calculated. The highest radioactivity concentration would probably be less than 4.6 uCi per drum based on the results of the highest concentration in RCP oil from an 860 MWE net unit owned by another utility (0.2 uCi Co-58, 0.3 uCi Co-60, 1.0 uCi Cs-134, and 3.1 uCi Cs-137 per drum). Assuming this worst case concentration were the average for all 1000 gallons incinerated per year at San Onofre, the dose to any organ of the maximum exposed individual (a child at the nearest residence located 1.3 miles NNW of the plant) was calculated to be 0.001 mrem/yr based on a X/Q of 1.2 E-6 sec/m^2 and a D/Q of 4.5 E-9 m^{-2}. This dose is 0.01% of the technical specification limit in Section 3/4.11.2.3.2 and is considered to be an insignificant contribution to dose via this pathway. The proposed technical specification would require calculations of doses associated with the incineration of each barrel, and would limit the accumulated dose during a calendar quarter or calendar year to less than 1% of 10 CFR 50, Appendix I limiting dose objectives. This is an appropriate small fraction of such limits for this source and is considered to be ALARA.

2. Will operation of the facility in accordance with this proposed amendment create the possiblity of a new or different kind of accident from any accident previousy evaluated?

   Response: No
   See above discussion.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in margin of safety?

   Response: No
   See above discussion.

Environmental Analysis

The Environmental Report estimates the following annual airborne releases will arise from the operation of San Onofre Units 2 and 3:

Noble Gases	8.6E+03 Curies/yr
Particulates	2.1E-01 Curies/yr
Radio-Iodine	2.0E-01 Curies/yr
Tritium	7.1E+02 Curies/yr
Total	9.3E+03 Curies/yr

Doses to a child resident at the mobile home park (1.3 miles NNW) resulting from the above expected airborne releases were calculated to be 1.48 mrem/yr whole body, 2.44 mrem/yr thyroid, and 1.86 mrem/yr skin with both units operating (environmental report - operating license stage, Table 5.2-1). The additional dose potentially contributed by the incineration of slightly contaminated oil is less than 1% of that currently calculated in the environmental report. Therefore, the inclusion of this new source does not significantly increase the environmental impact of the operations considered in the environmental report.

Safety and Significant Hazards Determination

Based on the above Safety and Environmental Analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

TABLE I

Results of Analyses of Oil Shipped from SONGS-1 on August 13, 1982

SAMPLE NUMBER	Cs-137 a. µCi/ml b. µCi/drum	Co-60 a. µCi/ml b. µCi/drum
1	a. 7.39E-8 b. 1.54E-2	<MDA <MDA
2	a. 1.2E-7 b. 2.5E-2	a. 8.4E-8 b. 1.75E-2
3	a. 5.2E-8 b. 1.08E-2	<MDA <MDA
4	a. 4.25E-8 b. 8.85E-3	<MDA <MDA

MDA VALUES:

Co-60 6.0E-8 µCi/ml
Cs-137 6.0E-8 µCi/ml

TABLE II*

*These data are the results of analyses of RCP oil from an 860 MWE net unit owned by another Utility and may be more representative of contamination levels to be expected in oil to be produced at SONGS 2/3.

Sample No.	Co-58 a) μCi/ml b) μCi/55 gal	Co-60 a) μCi/ml b) μCi/55 gal	Cs-134 a) μCi/ml b) μCi/55 gal	Cs-137 a) μCi/ml b) μCi/55 gal
2	<MDA	<MDA	a) 2.946E-6 b) 4.0516E-1	a) 9.636E-6 b) 2.0062
6	<MDA	a) 8.484E-7 b) 1.7664E-1	a) 9.803E-7 b) 2.041E-1	a) 3.334E-6 b) 6.914E-1
8	<MDA	<MDA	<MDA	a) 2.943E-6 b) 6.1273E-1
9	<MDA	<MDA	a) 9.500E-7 b) 1.9779E-1	a) 5.159E-6 b) 1.0741
11	a) 7.350E-7 b) 1.5303E-1	<MDA	a) 2.761E-6 b) 5.434E-1	a) 1.082E-5 b) 2.2527
14	<MDA	<MDA	<MDA	a) 2.130E-7 b) 4.4347E-2
17	<MDA	<MDA	<MDA	a) 6.199E-1 b) 1.2906E-1
18	<MDA	a) 1.125E-6 b) 2.3423E-1	a) 4.145E-7 b) 8.6299E-2	a) 2.242E-6 b) 5.0468E-1
20	<MDA	<MDA	<MDA	a) 4.575E-6 b) 9.5252E-1
21	<MDA	<MDA	a) 4.489E-6 b) 9.3461E-1	a) 1.501E-5 b) 3.1251
22	<MDA	<MDA	<MDA	a) 1.064E-6 b) 2.2152E-1
23	<MDA	<MDA	a) 3.408E-6 b) 7.0955E-1	a) 9.764E-6 b) 2.0329

TABLE II (Cont.)

Sample No.	Co-58 a) μCi/ml b) μCi/55 gal	Co-60 a) μCi/ml b) μCi/55 gal	Cs-134 a) μCi/ml b) μCi/55 gal	Cs-137 a) μCi/ml b) μCi/55 gal
25	<MDA	<MDA	<MDA	a) 7.042E-7 b) 1.4661E-1
26	<MDA	<MDA	a) 1.641E-6 b) 3.4166E-1	a) 6.639E-6 b) 1.3822
28	<MDA	a) 9.101E-7 b) 1.8948E-1	a) 4.181E-6 b) 8.7048E-1	a) 1.182E-5 b) 2.4609
29	<MDA	<MDA	a) 1.671E-6 b) 3.479E-1	a) 6.085E-6 b) 1.2669
30	<MDA	<MDA	<MDA	a) 1.267E-6 b) 2.6379E-1
31	<MDA	<MDA	<MDA	a) 1.463E-6 b) 3.046E-1
32	<MDA	<MDA	<MDA	a) 7.049E-7 b) 1.4676E-1
35	<MDA	<MDA	a) 1.087E-6 b) 2.2631E-1	a) 3.041E-6 b) 6.3314E-1
42	a) 5.668E-7 b) 1.1801E-1	<MDA	a) 7.647E-7 b) 1.5921E-1	a) 9.790E-7 b) 2.0383-1
45	<MDA	<MDA	<MDA	a) 1.126E-6 b) 2.3443E-1
47	<MDA	<MDA	a) 2.966E-6 b) 6.1752E-1	a) 7.992E-6 b) 1.6639
38	<MDA	a) 2.830E-7 b) 5.8921E-2	a) 1.270E-6 b) 2.6441E-1	a) 2.652E-6 b) 5.5215E-1

MDA Values

CO-58	1.76E-7	μCi/ml
CO-60	1.76E-7	μCi/ml
CS-134	1.76E-7	μCi/ml
CS-137	2.61E-7	μCi/ml

ATTACHMENT A

EXISTING SPECIFICATION

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate in unrestricted areas due to radioactive materials released in gaseous effluents from the site (see Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For all radioiodines, tritium and for all radioactive materials in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to radioiodines, tritium and radioactive materials in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) ($\mu Ci/ml$)[a]
A. Waste Gas Storage	P Each Tank Grab Sample	P Each Tank	Principle Gamma Emitters[g]	$1x10^{-4}$
B. Containment Purge 42 inch	P Each Purge[b,c]	P Each Purge[b]	Principle Gamma Emitters[g]	$1x10^{-4}$
			H-3	$1x10^{-6}$
8 inch	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g]	$1x10^{-4}$
			H-3	$1x10^{-6}$
C. 1. Condenser Evacuation System	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g]	$1x10^{-4}$
2. Plant Vent Stack	W[b,e]	W[b]	H-3	$1x10^{-6}$
D. All Release Types as listed in B and C above.	Continuous[f] Sampler	W[d] Charcoal Sample	I-131	$1x10^{-12}$
			I-133	$1x10^{-10}$
	Continuous[f] Sampler	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	$1x10^{-11}$
	Continuous[f] Sampler	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f] Sampler	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f] Monitor	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp\ (-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22×10^6 is the number of transformations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elaspsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a particular measurement system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank sample (as appropriate) rather than on an unverified theoretically predicted variance.

In calculating the LLD for a radionuclide determine by gamma ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing the capability of the measurement system and not as _a posteriori_ (after the fact) limit for a particular measurement.*

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*For a more complete discussion of the LLD, and other detection limits, see the following:

(1) HASL Procedures Manual, _HASL-300_ (revised annually).

(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" _Anal. Chem. 40_, 586-93 (1968).

(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report _ARH-2537_ (June 22, 1972).

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Analyses shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

c. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principle gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

ATTACHMENT B

PROPOSED SPECIFICATION

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate in unrestricted areas due to radioactive materials released in gaseous effluents from the site (see Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For all radioiodines, tritium and for all radioactive materials in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to radioiodines, tritium and radioactive materials in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) ($\mu Ci/ml$)[d]
A. Waste Gas Storage	P Each Tank Grab Sample	P Each Tank	Principle Gamma Emitters[g]	1×10^{-4}
B. Containment Purge 42 inch	P Each Purge[b,c]	P Each Purge[b]	Principle Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
8 inch	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
C. 1. Condenser Evacuation System	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
2. Plant Vent Stack	W[b,e]	W[b]	H-3	1×10^{-6}
D. All Release Types as listed in B and C above.	Continuous[f] Sampler	W[d] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[f] Sampler	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f] Sampler	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f] Sampler	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f] Monitor	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}
E. Auxiliary Boiler	Each 55 Gallon Drum of Oil Grab Sample	Each 55 Gallon Drum of Oil	Principal Gamma Emitters[g]	5×10^{-7}[h]

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp\ (-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volumne),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22×10^6 is the number of transformations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elaspsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a particular measurement system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank sample (as appropriate) rather than on an unverified theoretically predicted variance.

In calculating the LLD for a radionuclide determine by gamma ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of the measurement system and not as a posteriori (after the fact) limit for a particular measurement.*

*For a more complete discussion of the LLD, and other detection limits, see the following:
(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

TABLE NOTATION

b. Analyses shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

c. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principle gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

h. Sample LLD shall be μCi/ml of oil.

AUXILIARY BOILER

LIMITING CONDITION FOR OPERATION

3.11.2.7 Used oil, contaminated by radioactivity, may be incinerated in the auxiliary boiler provided releases do not exceed one percent (1%) of the limits set forth in Specification 3.11.2.3.

APPLICABILITY: At all times.

ACTION:

a. With releases from the auxiliary boiler exhaust exceeding the above limits, immediately suspend all further incineration of used oil which is contaminated by radioactivity for the duration of that calendar quarter or year corresponding to the dose limit which was exceeded.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 Surveillance requirements pertaining to releases in the auxiliary boiler exhaust are set forth in Specification 4.11.2.1.2.

Add the following to BASES Section B 3/4.11.2:

B 3/4.11.2.7 This specification provides assurance that particulate releases from this source will be an appropriate small fraction of the allowable particulate releases from all sources, yet provides the flexibility to safely dispose of used oil which has been slightly contaminated by radioactivity. Reasonable assurance is given that the dose to any organ of an individual in unrestricted areas will be less than 0.15 mrem/year from this source which is ALARA.

DESCRIPTION OF PROPOSED CHANGES
NPF-10-114 AND NPF-15-114 AND SAFETY ANALYSIS

This is a request to revise Technical Specification 4.6.1.6 "CONTAINMENT TENDONS".

Existing Specifications:

Units 2&3: See Attachments A and B

Proposed Specifications:

Units 2&3: See Attachments C and D

Description

The purpose of these changes is as follows:

1. The change to Paragraph 4.6.1.6 is for Unit 3 only and complies with the surveillance frequency required by Table 4.6-1 for tendon lift off force and tendon detensioning tests.

2. The change to the first sentence of Paragraph 4.6.1.6a clarifies the intent of the surveillance measurements as a means to verify structural integrity and removes an implied surveillance requirement that the lift-off force must be between the minimum and maximum values of the tolerance band. The tolerance band represents the normal range of variability in predictions of long term stress loss in each tendon over the life of the plant. Surveillance requirements are specifically and adequately covered at the end of this paragraph.

3. The remainder of the changes to Paragraph 4.6.1.6a provide clarification that the tolerance band values, particularly the upper one, are not to be treated as minimum and maximum limits but, rather, as surveillance guidelines. Specific actions associated with these guidelines are given at the end of the paragraph.

4. The last sentence of Paragraph 4.6.1.6a has been modified to clarify that Table 4.6-2 specifies the sample population.

5. The proposed change to Part c.3 is to require only the visual inspection of exposed concrete surfaces adjacent to the end anchorages of hoop tendons inspected instead of inspection of concrete surfaces adjacent to all end anchorages of tendons inspected. The existing conditions as shown in Detail 2 of Drawing 23005 do not permit the inspection of the concrete

surfaces adjacent to the U tendon end anchorages. The concrete is covered by 3/8-inch thick plates welded to the end anchorages and steel channels that are embedded in the concrete. The plates and channels were used as forms during construction of the containment base mat. Removal of the plates by such methods as grinding and flame cutting is not desirable because of potential damage to the adjacent concrete surfaces.

Concrete surfaces adjacent to hoop tendon and anchorages will be inspected per Technical Specification requirements. The prestress loads on hoop tendon end anchorages impose compressive and shear stresses on the buttress walls whereas the prestress loads on U tendon end anchorages impose only compressive stresses on the adjacent concrete. This occurs because the buttresses are projecting elements from the containment shell while the U tendon anchorage zone is in the same plane as the containment shell.

Safety Analysis

The proposed change discussed above shall be deemed to involve a significant hazards consideration if positive findings are made in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed changes to the Technical Specification requirements for tendon surveillance are for the purpose of clarification only and do not physically alter the surveillance program or the level of safety it provides. No physical change is involved to any part of the plant itself.

   For Part c.3 the elimination of visual inspection of concrete surfaces adjacent to U tendon end anchorages will not affect the structural integrity of the Containment Structure or the functions and operations of any equipment and systems. The compressive stresses in the concrete near the U tendon anchorage zone are low as shown in FSAR Table 3.8-2 and the prestress loads would preclude the formation of cracks in the adjacent concrete. All accident probabilities, consequences and scenarios remain bounded by existing analyses.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   As previously stated, the proposed change does not alter the physical plant configuration.

3. Will the operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No

As previously stated, no change will occur to the physical plant and associated margin of safety.

48 FR 14870 dated April 6, 1983 provided examples of amendments not likely to involve a significant hazards consideration. This proposed change is considered to be most similar to example (i) in that it involves changes to achieve consistency throughout the technical specifications, changes in nomenclature to clarify the intent of the requirements, and correction of an error in that it is not possible to visually inspect the concrete surfaces adjacent to U tendon anchorages because the concrete is covered by 3/8-inch metal plates welded to the end anchorages and steel channels that are embedded in the concrete.

Safety and Significant Hazards Determination

Based on the Safety Analysis, it is concluded that: (1) the proposed change does not involve a significant hazards consideration as defined by 10CFR50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the Station on the environment as described in the NRC Environmental Statement.

JKYann06a:npy

ATTACHMENT A

Existing Technical Specification

4.6.1.6 Containment Tendons

for

San Onofre Unit 2

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, perform an engineering evaluation of the containment to demonstrate its structural integrity within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 Containment Tendons The containment's structural integrity shall be demonstrated at the end of one, three and five years after the initial structural integrity test (ISIT) and every five years thereafter with the exception of tendon lift off force and tendon detensioning and material tests and inspections which shall be determined at the end of one, five and ten years following the ISIT and every ten years thereafter in accordance with Table 4.6-1. The structural integrity shall be demonstrated by:

a. Determining that tendons selected in accordance with Table 4.6-1 have a lift off force between the maximum and minimum values listed in Table 4.6-2 at the first year inspection. For subsequent inspections, for tendons and periodicities per Table 4.6-1, the maximum first year lift off forces shall be decreased by the amount X1 log t kips for U tendons and Y1 log t kips for hoop tendons and the minimum lift off forces shall be decreased by the amount X2 log t for U tendons and Y2 log t for hoop tendons where t is the time interval in years from initial tensioning of the tendon to the current testing date and the values X1, X2, Y1 and Y2 are in accordance with the values listed in Table 4.6-2 for the surveillance tendon. This test shall include essentially a complete detensioning of tendons selected in accordance with Table 4.6-1 in which the tendon is detensioned to determine if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to obtain a lift off force equal to +0, -5% of the prescribed upper limit. During retensioning of these tendons, the change in load and elongation shall be measured simultaneously at a minimum of three, approximately equally spaced, levels of force between the seating force and zero. If elongation corresponding to a spacific load differs by more than 5% from that recorded during installation of tendons, an investigation should

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be made to ensure that such difference is not related to wire failures or slip of wires in anchorages. If the lift off force of any one tendon in the total sample population lies between the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon shall be checked for their lift off force. If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. The tendon(s) shall be restored to the required level of integrity. More than one tendon below the predicted bounds out of the original sample population or the lift off force of a selected tendon lying below 90% of the prescribed lower limit is evidence of abnormal degradation of the containment structure.

b. Performing tendon detensioning and material tests and inspections of a previously stressed tendon wire or strand from one tendon of each group (hoop and U), and determining over the entire length of the removed wire or strand that:

   1. The tendon wires or strands are free of corrosion, cracks and damage.

   2. A minimum tensile strength value of 270 ksi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing a visual inspection of the following:

   1. Containment Surfaces - The structural integrity of the exposed accessible interior and exterior surfaces of the containment shall be determined during the shutdown for, and prior to, each Type A containment leakage rate test (Specification 4.6.1.2) by a visual inspection of these surfaces and verifying no apparent changes in appearance or other abnormal degradation (e.g., widespread cracking, spalling and/or grease leakage).

   2. End Anchorages - The structural integrity of the end anchorages (e.g., bearing plates, stressing washers, shims, wedges and anchorheads) of all tendons inspected pursuant to Specification 4.6.1.6a shall be demonstrated by inspection that no apparent changes have occurred in the visual appearance of the end anchorage.

   3. Concrete Surfaces - The structural integrity of the concrete surfaces adjacent to the end anchorages of tendons inspected

SURVEILLANCE REQUIREMENTS (Continued)

pursuant to Specification 4.6.1.6a shall be demonstrated by visual examination of the crack patterns to verify no abnormal material behavior.

d. Verifying the OPERABILITY of the sheathing filler grease by the following:

1. No significant voids (in excess at 5% of the net duct volume), or the presence of free water, within the grease filler material, taking into account temperature variations.

2. No significant changes have occurred in the physical appearance of the sheathing filler grease.

3. Complete grease coverage exists for the anchorage system.

4. Chemical properties are within the tolerance limits specified by the sheathing filler grease manufacturer.

ATTACHMENT B

Existing Technical Specification

4.6.1.6 Containment Tendons

for

San Onofre Unit 3

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, perform an engineering evaluation of the containment to demonstrate its structural integrity within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 Containment Tendons The containment's structural integrity shall be demonstrated at the end of one, three and five years after the initial structural integrity test (ISIT) and every five years thereafter with the exception of tendon lift off force and tendon detensioning and material tests and inspections which shall be determined at the end of one, five and ten years following the ISIT and every ten years thereafter in accordance with Table 4.6-1. The structural integrity shall be demonstrated by:

a. Determining that tendons selected in accordance with Table 4.6-1 have a lift off force between the maximum and minimum values listed in Table 4.6-2 at the first year inspection. For subsequent inspections, for tendons and periodicities per Table 4.6-1, the maximum first year lift off forces shall be decreased by the amount X1 log t kips for U tendons and Y1 log t kips for hoop tendons and the minimum lift off forces shall be decreased by the amount X2 log t for U tendons and Y2 log t for hoop tendons where t is the time interval in years from initial tensioning of the tendon to the current testing date and the values X1, X2, Y1 and Y2 are in accordance with the values listed in Table 4.6-2 for the surveillance tendon. This test shall include essentially a complete detensioning of tendons selected in accordance with Table 4.6-1 in which the tendon is detensioned to determine if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to obtain a lift off force equal to +0, -5% of the prescribed upper limit. During retensioning of these tendons, the change in load and elongation shall be measured simultaneously at a minimum of three, approximately equally spaced, levels of force between the seating force and zero. If elongation corresponding to a specific load differs by more than 5% from that recorded during installation of tendons, an investigation should be made to ensure that such

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difference is not related to wire failures or slip of wires in anchorages. If the lift off force of any one tendon in the total sample population lies between the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon shall be checked for their lift off force. If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. The tendon(s) shall be restored to the required level of integrity. More than one tendon below the predicted bounds out of the original sample population or the lift off force of a selected tendon lying below 90% of the prescribed lower limit is evidence of abnormal degradation of the containment structure.

b. Performing tendon detensioning and material tests and inspections of a previously stressed tendon wire or strand from one tendon of each group (hoop and U), and determining that over the entire length of the removed wire or strand that:

   1. The tendon wires or strands are free of corrosion, cracks and damage.

   2. A minimum tensile strength value of 270 ksi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing a visual inspection of the following:

   1. Containment Surfaces - The structural integrity of the exposed accessible interior and exterior surfaces of the containment shall be determined during the shutdown for, and prior to, each Type A containment leakage rate test (Specification 4.6.1.2) by a visual inspection of these surfaces and verifying no apparent changes in appearance or other abnormal degradation (e.g., widespread cracking, spalling and/or grease leakage).

   2. End Anchorages - The structural integrity of the end anchorages (e.g., bearing plates, stressing washers, shims, wedges and anchorheads) of all tendons inspected pursuant to Specification 4.6.1.6a shall be demonstrated by inspection that no apparent changes have occurred in the visual appearance of the end anchorage.

   3. Concrete Surfaces - The structural integrity of the concrete surfaces adjacent to the end anchorages of tendons inspected

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SURVEILLANCE REQUIREMENTS (Continued)

pursuant to Specification 4.6.1.6a shall be demonstrated by visual examination of the crack patterns to verify no abnormal material behavior.

d. Verifying the OPERABILITY of the sheathing filler grease by the following:

1. No significant voids (in excess of 5% of the net duct volume), or the presence of free water, within the grease filler material, taking into account temperature variations.

2. No significant changes have occurred in the physical appearance of the sheathing filler grease.

3. Complete grease coverage exists for the anchorage system.

4. Chemical properties are within the tolerance limits specified by the sheathing filler grease manufacturer.

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ATTACHMENT C

Proposed Technical Specification

4.6.1.6 Containment Tendons

for

San Onofre Unit 2

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, perform an engineering evaluation of the containment to demonstrate its structural integrity within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 Containment Tendons The containment's structural integrity shall be demonstrated at the end of one, three and five years after the initial structural integrity test (ISIT) and every five years thereafter with the exception of tendon lift off force and tendon detensioning and material tests and inspections which shall be determined at the end of one, five and ten years following the ISIT and every ten years thereafter in accordance with Table 4.6-1. The structural integrity shall be demonstrated by:

a. Determining the lift off force of tendons selected in accordance with Table 4.6-1 and comparing this force with the tolerance band values listed in Table 4.6-2 at the first year inspection. For subsequent inspections, for tendons and periodicities per Table 4.6-1, the upper tolerance band value for first year lift off forces shall be decreased by the amount X1 log t kips for U tendons and Y1 log t kips for hoop tendons and the lower tolerance band value for lift off forces shall be decreased by the amount X2 log t for U tendons and Y2 log t for hoop tendons where t is the time interval in years from initial tensioning of the tendon to the current testing date and the values X1, X2, Y1 and Y2 are in accordance with the values listed in Table 4.6-2 for the surveillance tendon. This test shall include essentially a complete detensioning of tendons selected in accordance with Table 4.6-1 in which the tendon is detensioned to determine if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to obtain a lift off force equal to +0, -5% of the prescribed upper tolerance band value. During retensioning of these

tendons, the change in the load and elongation shall be measured simultaneously at a minimum of three, approximately equally spaced, levels of force between the seating force and zero. If elongation corresponding to a specific load differs by more than 5% from that recorded during installation of tendons, an investigation should be made to ensure that such difference is not related to wire failures or slip of wires in anchorages. If the lift off force of any one tendon in the total sample population lies between the prescribed lower tolerance band value and 90% of the prescribed lower tolerance band value two tendons, one on each side of this tendon shall be checked for their lift off force. If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. The tendon(s) shall be retensioned such that the lift off force is equal to +0, -5% of the prescribed upper tolerance band value. The following lift off force measurement results are considered to be evidence of abnormal degradation of the containment structure:

1. More than one tendon from Table 4.6-2 or adjacent tendons, below the lower tolerance band value.

2. The lift off force of a selected tendon from Table 4.6-2 lying below 90% of the prescribed lower tolerance band value.

b. Performing tendon detensioning and material tests and inspections of a previously stressed tendon wire or strand from one tendon of each group (hoop and U), and determining over the entire length of the removed wire or strand that:

1. The tendon wires or strands are free of corrosion, cracks and damage.

2. A minimum tensile strength value of 270 ksi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing a visual inspection of the following:

1. Containment Surfaces - The structural integrity of the exposed accessible interior and exterior surfaces of the containment shall be determined during the shutdown for, and prior to, each Type A containment leakage rate test (Specification 4.6.1.2) by a visual inspection of these surfaces and verifying no apparent changes in appearance or other abnormal degradation (e.g., widespread cracking, spalling and/or grease leakage).

SURVEILLANCE REQUIREMENTS (Continued)

2. End Anchorages - The structural integrity of the end anchorages (e.g., bearing plates, stressing washers, shims, wedges and anchorheads) of all tendons inspected pursuant to Specification 4.6.1.6a shall be demonstrated by inspection that no apparent changes have occurred in the visual appearance of the end anchorage.

3. Concrete Surfaces - The structural integrity of the exposed concrete surfaces adjacent to the end anchorages of hoop tendons inspected pursuant to Specification 4.6.1.6a shall be demonstrated by visual examination of the crack patterns to verify no abnormal material behavior.

d. Verifying the OPERABILITY of the sheathing filler grease by the following:

1. No significant voids (in excess of 5% of the net duct volume) or the presence of free water, within the grease filler material, taking into account temperature variations.

2. No significant changes have occurred in the physical appearance of the sheathing filler grease.

3. Complete grease coverage exists for the anchorage system.

4. Chemical properties are within the tolerance limits specified by the sheathing filler grease manufacturer.

ATTACHMENT D

Proposed Technical Specification

4.6.1.6 Containment Tendons

for

San Onofre Unit 3

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, perform an engineering evaluation of the containment to demonstrate its structural integrity within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 Containment Tendons The containment's structural integrity shall be demonstrated at the end of one, three and five years after the initial structural integrity test (ISIT) and every five years thereafter with the exception of tendon lift off force and tendon detensioning and material tests and inspections which shall be determined at the end of one and five years following the ISIT and every ten years thereafter in accordance with Table 4.6-1. The structural integrity shall be demonstrated by:

a. Determining the lift off force of tendons selected in accordance with Table 4.6-1 and comparing this force with the tolerance band values listed in Table 4.6-2 at the first year inspection. For subsequent inspections, for tendons and periodicities per Table 4.6-1, the upper tolerance band value for first year lift off forces shall be decreased by the amount X1 log t kips for U tendons and Y1 log t kips for hoop tendons and the lower tolerance band value for lift off forces shall be decreased by the amount X2 log t for U tendons and Y2 log t for hoop tendons where t is the time interval in years from initial tensioning of the tendon to the current testing date and the values X1, X2, Y1 and Y2 are in accordance with the values listed in Table 4.6-2 for the surveillance tendon. This test shall include essentially a complete detensioning of tendons selected in accordance with Table 4.6-1 in which the tendon is detensioned to determine if any wires or strands are broken or damaged. Tendons found acceptable during this test shall be retensioned to obtain a lift off force equal to +0, -5% of the prescribed upper tolerance band value. During retensioning of these

tendons, the change in the load and elongation shall be measured simultaneously at a minimum of three, approximately equally spaced, levels of force between the seating force and zero. If elongation corresponding to a specific load differs by more than 5% from that recorded during installation of tendons, an investigation should be made to ensure that such difference is not related to wire failures or slip of wires in anchorages. If the lift off force of any one tendon in the total sample population lies between the prescribed lower tolerance band value and 90% of the prescribed lower tolerance band value two tendons, one on each side of this tendon shall be checked for their lift off force. If both of these adjacent tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. The tendon(s) shall be retensioned such that the lift off force is equal to +0, -5% of the prescribed upper tolerance band value. The following lift off force measurement results are considered to be evidence of abnormal degradation of the containment structure:

1. More than one tendon from Table 4.6-2 or adjacent tendons, below the lower tolerance band value.

2. The lift off force of a selected tendon from Table 4.6-2 lying below 90% of the prescribed lower tolerance band value.

b. Performing tendon detensioning and material tests and inspections of a previously stressed tendon wire or strand from one tendon of each group (hoop and U), and determining over the entire length of the removed wire or strand that:

   1. The tendon wires or strands are free of corrosion, cracks and damage.

   2. A minimum tensile strength value of 270 ksi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing a visual inspection of the following:

   1. Containment Surfaces - The structural integrity of the exposed accessible interior and exterior surfaces of the containment shall be determined during the shutdown for, and prior to, each Type A containment leakage rate test (Specification 4.6.1.2) by a visual inspection of these surfaces and verifying no apparent changes in appearance or other abnormal degradation (e.g., widespread cracking, spalling and/or grease leakage).

SURVEILLANCE REQUIREMENTS (Continued)

2. End Anchorages - The structural integrity of the end anchorages (e.g., bearing plates, stressing washers, shims, wedges and anchorheads) of all tendons inspected pursuant to Specification 4.6.1.6a shall be demonstrated by inspection that no apparent changes have occurred in the visual appearance of the end anchorage.

3. Concrete Surfaces - The structural integrity of the exposed concrete surfaces adjacent to the end anchorages of hoop tendons inspected pursuant to Specification 4.6.1.6a shall be demonstrated by visual examination of the crack patterns to verify no abnormal material behavior.

d. Verifying the OPERABILITY of the sheathing filler grease by the following:

1. No significant voids (in excess of 5% of the net duct volume) or the presence of free water, within the grease filler material, taking into account temperature variations.

2. No significant changes have occurred in the physical appearance of the sheathing filler grease.

3. Complete grease coverage exists for the anchorage system.

4. Chemical properties are within the tolerance limits specified by the sheathing filler grease manufacturer.

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DESCRIPTION OF PROPOSED CHANGES NPF-10-119 AND NPF-15-119 AND SAFETY ANALYSIS

This is a request to revise Technical Specification Basis 3/4.2.3, "Azimuthal Power Tilt - T_q."

Existing Specifications:

Units 2 and 3: See Attachment "A"

Proposed Specifications:

Units 2 and 3: See Attachment "B"

Description:

This change clarifies the Basis of LCO 3/4.2.3, "Azimuthal Power Tilt-T_q." The current description of the Azimuthal Power Tilt Bases creates the false impression that CPC's provide an input to COLSS for calculation of azimuthal power tilt and that COLSS does not calculate azimuthal power tilt below 20% of rated thermal power. In reality, COLSS calculates azimuthal power tilt based on its input from the fixed incore neutron detectors and has no connection with the CPC's. Additionally, this calculation is performed at all power levels although it is unreliable below 20% rated thermal power.

Safety Analysis:

The proposed change discussed above shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   Operation of the facility will be unaffected since the proposed change is merely a clarified explanation in the bases section of the technical specifications. The change does not increase the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

Safety Analysis (continued):

As explained above, operation of the facility will be unaffected. The change does not create the possibility of a new or different kind of accident.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   As explained above, operation of the facility will be unaffected. The change does not involve a reduction in a margin of safety.

The proposed revision of the Azimuthal Power Tilt Bases is similar to example (i) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864, dated April 6, 1983.

Safety and Significant Hazards Determination:

Based on the Safety Analysis, it is concluded that: (1) the proposed change does not involve a significant hazards consideration as defined by 10CFR50.92; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and, (3) this action will not result in a condition which significantly alters the impact of the Station on the environment as described in the NRC Final Environmental Statement.

ATTACHMENT B

(Proposed Specification)

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

This LCO does not apply below 20% rated thermal power for two reasons:

(1) The incore neutron detectors are inaccurate at low core power levels due to the poor signal-to-noise ratio which they experience. The resultant COLSS AZIMUTHAL POWER TILT is unreliable since COLSS uses the incore neutron detector signals to perform this calculation.

(2) The CPC's assume a minimum core power of 20% rated thermal power. When actual power is below this level the core is operating further from thermal limits and the resultant CPC-calculated DNBR and LPD trips are highly conservative.

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt}=1+T_q g \cos(0-0_o)$$

where:

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

0 is the azimuthal core location

0_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

ATTACHMENT A

(Existing Specification)

3/4.2.2 PLANAR RADIAL PEAKING FACTORS

Limiting the values of the PLANAR RADIAL PEAKING FACTORS (F_{xy}^{c}) used in the COLSS and CPCs to values equal to or greater than the measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^{m}) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured PLANAR RADIAL PEAKING FACTORS. A minimum core power at 20% of RATED THERMAL POWER is assumed in determining the PLANAR RADIAL PEAKING FACTORS. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The periodic surveillance requirements for determining the measured PLANAR RADIAL PEAKING FACTORS provides assurance that the PLANAR RADIAL PEAKING FACTORS used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured PLANAR RADIAL PEAKING FACTORS after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. A minimum core power of 20% of RATED THERMAL POWER is assumed by the CPCs in its input to COLSS for calculation of AZIMUTHAL POWER TILT. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q\, g \cos(\theta - \theta_0)$$

where:

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AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minumum DNBR limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPC's which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings.

DESCRIPTION OF PROPOSED CHANGES NPF-10-123 AND NPF-15-123
AND SAFETY ANALYSIS

This is a request to revise Technical Specification Section 2.0 Bases, DNBR-Low.

Existing Specifications:

Units 2 and 3: See Attachment "A"

Proposed Specifications:

Units 2 and 3: See Attachment "B"

Description:

This change makes clarifying and editorial changes to the Technical Specification Section 2 Bases, "Departure from Nucleate Boiling Ratio-Low." The specific changes and reasons are:

1. The section heading is changed from "DNBR-Low" to "Departure from Nucleate Boiling Ratio (DNBR)-Low." This is strictly an editorial change.

2. Two sentences in the first paragraph indicating that a DNBR-Low trip occurs if pressurizer pressure drops below 1825 psia are deleted. This information is included in the later table of trip values. There is no reason to emphasize the low pressurizer pressure trip over any of the other DNBR-Low trip parameters.

3. The paragraph above the table of trip values was changed to emphasize that the limits specified apply only to the DNBR algorithm used for the DNBR-Low trip. The actual plant limiting condition for operation for a specific parameter may be more restrictive than the value used for the CPC DNBR-Low trip.

4. Parameter (i) in the table of trip values is changed from "Quality Margin-Low $<$ 0" to Hot Leg Quality 0 (no net quality)." Hot leg quality is the correct trip variable. Quality Margin is used in the DNBR algorithm but is not a trip variable.

Safety Analysis:

The proposed change discussed above shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with the proposed change involve a significant increase in the probability or

Safety Analysis (continued):

consequences of an accident previously evaluated?

Response: No

The changes proposed are editorial for clarity and do not affect the probability or consequences of an accident.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The changes proposed are editorial for clarity and do not create the possibility of a new or different kind of accident.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The changes proposed are editorial for clarity and do not affect any margin of safety.

   The proposed revision of the DNBR Bases is similar to example (i) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983, in that it is essentially administrative in nature.

Safety and Significant Hazards Determination:

Based on the Safety Analysis, it is concluded that: (1) the proposed change does not involve a significant hazards consideration as defined by 10CFR50.92; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and, (3) this action will not result in a condition which significantly alters the impact of the Station on the environment as described in the NRC Final Environmental Statement.

ATTACHMENT B

(Proposed Specification)

Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

Departure from Nucleate Boiling Ratio (DNBR) - Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences.

The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.20 such that the decrease in actual core

DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the parametric envelope indicated below which is typically less restrictive than plant specific limiting conditions for operation. Operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	$\geq$ 495°F
b.	RCS Cold Leg Temperature-High	$\leq$ 580°F
c.	Axial Shape Index-Positive	< +0.5
d.	Axial Shape Index-Negative	$\geq$ -0.5
e.	Pressurizer Pressure-Low	$\geq$ 1825 psia
f.	Pressurizer Pressure-High	$\leq$ 2375 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Hot Leg Quality	0 (no net quality)

The DNBR Trip setpoint in CPC and COLSS is 1.19. The values of the penalty factors BERR1 (CPC) and EPOL2 (COLSS) may be adjusted to implement requirements for tripping at other values of DNBR. The following formula is used to adjust the CPC addressable constant BERR1:

$$BERR1_{new} = BERR1_{old} \left[1 + \Delta DNBR(\%) * \left|\frac{d(\%\ POL)}{d\ (\%\ DNBR)}\right| * 0.01\right]$$

where:

$BERR1_{new}$ = new required value of BERR1,

$BERR1_{old}$ = present implemented value of BERR1,

$\Delta DNBR(\%)$ = percent increase in DNBR trip setpoint requirement,

$|d(\%\ POL)/d(\%\ DNBR)|$ = The absolute value of the most adverse derivative of percent POL with respect to percent DNBR as reported in CEN-184(S)-P.

Similarly, for the COLSS addressable constant EPOL2:

$$EPOL2_{new} = \left(1 + \Delta DNBR(\%) * \left|\frac{d(\%\ POL)}{d\ (\%\ DNBR)}\right| * 0.01\right) * (1 + EPOL2_{old}) - 1.0$$

where:

$EPOL2_{new}$ = new required value of EPOL2,

$EPOL2_{old}$ = present implemented value of EPOL2,

and the other terms are as previously defined.

ATTACHMENT A

(Existing Specification)

Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1825 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.20 such that the decrease in actual core

DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	$\geq$ 495°F
b.	RCS Cold Leg Temperature-High	$\leq$ 580°F
c.	Axial Shape Index-Positive	$<$ +0.5
d.	Axial Shape Index-Negative	$\geq$ -0.5
e.	Pressurizer Pressure-Low	$\geq$ 1825 psia
f.	Pressurizer Pressure-High	$\leq$ 2375 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	$<$ 0

The DNBR Trip setpoint in CPC and COLSS is 1.19. The values of the penalty factors BERR1 (CPC) and EPOL2 (COLSS) may be adjusted to implement requirements for tripping at other values of DNBR. The following formula is used to adjust the CPC addressable constant BERR1:

$$BERR1_{new} = BERR1_{old} \left[1 + \Delta DNBR(\%) * \left|\frac{d(\%\ POL)}{d\ (\%\ DNBR)}\right| * 0.01\right]$$

where:

$BERR1_{new}$ = new required value of BERR1,

$BERR1_{old}$ = present implemented value of BERR1,

$\Delta DNBR(\%)$ = percent increase in DNBR trip setpoint requirement,

$|d(\%\ POL)/d(\%\ DNBR)|$ = The absolute value of the most adverse derivative of percent POL with respect to percent DNBR as reported in CEN-184(S)-P.

Similarly, for the COLSS addressable constant EPOL2:

$$EPOL2_{new} = \left(1 + \Delta DNBR(\%) * \left|\frac{d(\%\ POL)}{d\ (\%\ DNBR)}\right| * 0.01\right) * (1 + EPOL2_{old}) - 1.0$$

where:

$EPOL2_{new}$ = new required value of EPOL2,

$EPOL2_{old}$ = present implemented value of EPOL2,

and the other terms are as previously defined.

DESCRIPTION OF PROPOSED CHANGES NPF-10-125 AND NPF-15-125
AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.6.2.2, "Iodine Removal System."

Existing Specifications:

Unit 2: See Attachment "A"
Unit 3: See Attachment "B"

Proposed Specifications:

Unit 2: See Attachment "C"
Unit 3: See Attachment "D"

Description:

The Unit 2 version of LCO 3.6.2.2 contains incorrect wording regarding spray chemical solution temperature requirements. The existing LCO calls for "...a minimum solution temperature between 82°F and 88°F..." The Unit 3 Technical Specifications have been corrected to require "...a solution temperature between 82°F and 104°F..." The attached proposed change corrects the Unit 2 Technical Specifications by substituting the updated Unit 3 temperature requirements for the incorrect existing requirements. Additionally, the proposed change provides the indicated tank levels which correspond to the minimum required usable spray chemical solution volume. Since a lower tank level is required when the low level cutout is bypassed (making the tank's entire contents usable) two different indicated levels are provided:

(1) 85% with low level cutout in service.
(2) 67% with low level cutout bypassed.

This second change is applicable to both Units 2 and 3.

Safety Analysis:

The proposed change discussed above shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   Operation of the facility will be unaffected since the proposed change is merely a clarification of the requirements for volume and temperature of the spray chemical solution. The change does not

Safety Analysis (continued):

involve an increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   As explained above, operation of the facility will be unaffected. The change does not create the possibility of a new or different kind of accident.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   As explained above, operation of the facility will be unaffected. The change does not involve a reduction in a margin of safety.

The proposed revision of the Iodine Removal System requirements is similar to example (iv) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864, dated April 6, 1983.

Safety and Significant Hazards Determination:

Based on the Safety Analysis, it is concluded that: (1) the proposed change does not involve a significant hazards consideration as defined by 10CFR50.92; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and, (3) this action will not result in a condition which significantly alters the impact of the Station on the environment as described in the NRC Final Environmental Statement.

ATTACHMENT C

(Proposed Specification)

IODINE REMOVAL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The iodine removal system shall be OPERABLE with:

a. A spray additive tank containing a minimum usable solution volume of 1456 gallons* of between 40% and 44% by weight NaOH solution with a solution temperature between 82°F and 104°F and

b. Two spray chemical addition pumps each capable of adding NaOH solution from the chemical addition tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the iodine removal system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the iodine removal system to OPERABLE status within the next 48 hours or be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The iodine removal system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the NaOH solution temperature.

b. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

d. At least once per 18 months, during shutdown, by verifying that (1) each automatic valve in the flow path actuates to its correct position, and (2) that each spray chemical addition pump starts automatically on a Containment Spray Actuation test signal.

e. At least once per 5 years by verifying a minimum solution flow rate of 20 gpm through all piping sections from the spray additive tank to the suction at the containment spray pumps.

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*If the spray additive tank low level cutout is in service, the minimum indicated volume required is 85%. If the cutout is bypassed, an indicated level of 67% is sufficient to provide 1456 gallons of usable solution.

ATTACHMENT D
(Proposed Specifications)

CONTAINMENT SYSTEMS

IODINE REMOVAL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The iodine removal system shall be OPERABLE with:

a. A spray additive tank containing a minimum usable solution volume of 1456 gallons* of between 40% and 44% by weight NaOH solution with a solution temperature between 82°F and 104°F and

b. Two spray chemical addition pumps each capable of adding NaOH solution from the chemical addition tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the iodine removal system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the iodine removal system to OPERABLE status within the next 48 hours or be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The iodine removal system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the NaOH solution temperature.

b. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

d. At least once per 18 months, during shutdown, by verifying that (1) each automatic valve in the flow path actuates to its correct position on and (2) that each spray chemical addition pump starts automatically on a Containment Spray Actuation test signal.

e. At least once per 5 years by verifying a minimum solution flow rate of 20 gpm through all piping sections from the spray additive tank to the suction at the containment spray pumps.

*If the spray additive tank low level cutout is in service, the minimum indicated volume required is 85%. If the cutout is bypassed, an indicated level of 67% is sufficient to provide 1456 gallons of usable solution.

ATTACHMENT A

(Existing Specification)

IODINE REMOVAL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The iodine removal system shall be OPERABLE with:

a. A spray additive tank containing a minimum solution volume of 1456 gallons of between 40 and 44% by weight NaOH solution with a minimum solution temperature between 82°F and 88°F and

b. Two spray chemical addition pumps each capable of adding NaOH solution from the chemical addition tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the iodine removal system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the iodine removal system to OPERABLE status within the next 48 hours or be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The iodine removal system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the NaOH solution temperature.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 6 months by:

   1. Verifying the contained solution volume in the tank, and

   2. Verifying the concentration of the NaOH solution by chemical analysis.

d. At least once per 18 months, during shutdown, by verifying that (1) each automatic valve in the flow path actuates to its correct position and (2) that each spray chemical addition pump starts automatically on a Containment Spray Actuation test signal.

e. At least once per 5 years by verifying a minimum solution flow rate of 20 gpm through all piping sections from the spray additive tank to the suction at the containment spray pumps.

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ATTACHMENT B

(Existing Specification)

CONTAINMENT SYSTEMS

IODINE REMOVAL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The iodine removal system shall be OPERABLE with:

a. A spray additive tank containing a minimum solution volume of 1456 gallons of between 40 and 44% by weight NaOH solution with a solution temperature between 82°F and 104°F and

b. Two spray chemical addition pumps each capable of adding NaOH solution from the chemical addition tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the iodine removal system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the iodine removal system to OPERABLE status within the next 48 hours or be in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The iodine removal system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the NaOH solution temperature.

b. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

d. At least once per 18 months, during shutdown, by verifying that (1) each automatic valve in the flow path actuates to its correct position and (2) that each spray chemical addition pump starts automatically on a Containment Spray Actuation test signal.

e. At least once per 5 years by verifying a minimum solution flow rate of 20 gpm through all piping sections from the spray additive tank to the suction at the containment spray pumps.

ISSUED TO A
CONTROLLED LOCATION

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DESCRIPTION OF PROPOSED CHANGES NPF-10-127 AND NPF-15-127
AND SAFETY ANALYSIS

This is a request to add Section 3/4.7.10, "Emergency Chilled Water System" to the Technical Specifications for San Onofre Nuclear Generating Station, Units 2 and 3.

Description

The San Onofre Units 2 and 3 Technical Specifications currently address the Emergency Chilled Water System (ECWS) through the definition of OPERABILITY, Section 1.17, as a required support system. The above addition is requested in order to delineate the Limiting Condition for Operation (LCO) requirements for the ECWS. Technical Specification 3/4.7.10 has been developed through consideration of the support function of the ECWS, existing Technical Specification requirements and the consequences of ECWS inoperability during normal and emergency plant operating conditions.

The ECWS, in conjunction with the respective emergency HVAC units, is required (according to Technical Specification definition 1.17) to provide heat removal in maintaining the various Engineered Safety Feature (ESF) room space design temperatures below the associated equipment qualification limits for the range of design basis accident conditions. The normal HVAC system is redundant to the emergency HVAC system in maintaining the space design conditions of required safety systems during normal operating conditions and design basis accident conditions not involving seismic events or loss of offsite power. A 7 day action requirement is proposed for a single ECWS out of service, based on the high reliability of offsite power and availability of the normal HVAC system (the normal HVAC system contains two 100 percent redundant chillers, with one chiller normally in continuous operation and the other in standby). This requirement is consistent with Section 3/4.7.5 governing the control room emergency air cleanup system. Action requirements are provided to ensure OPERABILITY of the vital bus inverters and emergency battery chargers, by verifying within one hour that the normal HVAC system is providing space cooling to the vital power distribution rooms (loss of space cooling will not result in loss of vital bus inverter or emergency battery charger function in less than 75 minutes). Probablistic risk assessment studies at similar plants have shown that seismic events and events involving fire are the major contributors to accident risk. Therefore, an ACTION requirement is provided to establish within 8 hours operability of the safe shutdown systems which do not depend on the inoperable ECWS. The 8 hour period provides a reasonable time in which to establish OPERABILITY of this complement of key safety systems. This requirement ensures that a functional train of safe shutdown equipment is available to put the plant in a safe, stable condition for the most probable abnormal operational occurrences. An action requirement of 24 hours is provided to establish operability of the remaining required safety systems which do not depend on the inoperable ECWS. This period permits completion of maintenance activities and/or activation of third-of-a-kind components to ensure that an OPERABLE train of safety systems is available to mitigate the range of design basis events during the remainder of the Action.

As discussed in FSAR Appendix 3B on shared systems, the ECWS is started by safety signals generated by either unit. The proposed Technical Specification will be the same for both units.

The proposed Technical Specification addresses plant operation in Modes 1 through 4. Operation in Modes 5 and 6 is to be addressed through the definition of operability, Technical Specification Section 1.17.

Proposed Technical Specifications

Technical Specification 3/4.7.10 for Units 2 and 3 is provided as Attachment A.

Safety Analysis

The proposed change discussed above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated in the FSAR?

   Response: No

   The proposed Technical Specification prescribes Limiting Condition for Operation requirements based on and consistant with existing Technical Specifications as described above. The proposed Technical Specification does not involve a change in the plant configuration. The proposed Technical Specification delineates the requirements governing ECWS operability, in order to facilitate plant operation consistent with that assumed in the FSAR accident analysis. Therefore, operation under Technical Specification 3/4.7.10 will not involve a significant increase in the probability or consequences of any previously evaluated accident.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or differenct kind of accident from any accident previously evaluated in the FSAR?

   Response: No

   The consequences of plant operation under the proposed Technical Specification remain bounded by existing FSAR analyses. The proposed Technical Specification does not involve a change in the plant configuration. Therefore, the proposed Technical Specification does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The margin of safety as defined by existing Technical Specifications is maintained by this proposed addition. The proposed Technical Specification prescribes Limiting Conditions for Operation consistent with existing Technical Specifications. The consequences of plant operation under the proposed Technical Specification remain bounded by existing analyses. Therefore, the proposed Technical Specification does not involve a significant reduction in a margin of safety.

The proposed Technical Specification addressing the emergency chilled water system, is pertinent to Section 50.91, example (ii) of 48 FR 14870 dated April 6, 1983, entitled: Ammendments That Are Considered Not Likely To Involve a Significant Hazards Consideration, in that the change constitutes an additional limitation /restriction/ control not presently included in the Technical Specifications.

<u>Safety and Significant Hazards Determination</u>

Based on the Safety Analysis it is concluded that:

1. The proposed change does not involve a significant hazards consideration as defined by 10 CFR 50.92; and,

2. There is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and,

3. This action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT "A"

3/4.7.10 EMERGENCY CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10 Two independent emergency chilled water systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With only one emergency chilled water system OPERABLE, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With only one emergency chilled water system OPERABLE:

1. Within 1 hour verify that the normal HVAC system is providing space cooling to the vital power distribution rooms that depend on the inoperable emergency chilled water system for space cooling, and

2. Within 8 hours establish OPERABILITY of the safe shutdown systems which do not depend on the inoperable emergency chilled water system (one train each of boration, pressurizer heaters and auxiliary feedwater per Sections 3/4.1.2.2, 3/4.4.3 and 3/4.7.1.2, respectively) and

3. Within 24 hours establish OPERABILITY of all required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE emergency chilled water system for space cooling.

If these conditions are not satisfied within the specified time, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.10 Each of the above required emergency chilled water systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each manual valve servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position and,

EMERGENCY CHILLED WATER SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months by verifying that: each power operated or automatic valve servicing safety-related equipment actuates to its correct position and each chilled water pump starts automatically on a TGIS, CRIS, SIAS and, with irradiated fuel in the storage pool, FHIS.

BASES

3/4.7.10 Emergency Chilled Water System

The OPERABILITY of the emergency chilled water system ensures that space cooling capacity is available for continued operation of safety-related equipment during accident conditions. The redundant cooling capacity of these systems is consistent with single failure criteria and the assumptions used in the accident analyses. A 7 day ACTION requirement is specified in the event of emergency chilled water system inoperability, based on the high reliability of offsite power and availability of the normal HVAC system. Further Actions (b.1, b.2 and b.3) are specified regarding the status of systems and equipment which are unit specific and common. These Actions are not intended to apply to the status of systems and equipment specific to the other Unit. When one emergency chilled water system is inoperable, there is an ACTION requirement to verify within 1 hour that the normal HVAC system is providing space cooling to the vital power distribution rooms that are served by the inoperable emergency chilled water system. Availability of the normal HVAC system permits the vital bus inverters and emergency battery chargers to continue to be considered OPERABLE by ensuring that environmental qualification limits are not exceeded (the loss of space cooling to the vital power distribution rooms will not result in loss of vital bus inverter or emergency battery charger function in less than 75 minutes). The requirement to establish OPERABILITY of all required systems, subsystems, trains, components and devices, that depend on the remaining emergency chilled water system for space cooling, is intended to provide assurance that an OPERABLE train of safety-related equipment is available to meet the requirements of the range of design basis events. The requirement to establish OPERABILITY of the minimum safe shutdown systems ensures that at least one train of safety systems is available to place the plant in a safe, stable condition in the event of transients such as loss of offsite power, a safe shutdown earthquake, or design basis fire prior to establishment of OPERABILITY of the remaining required safety systems. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate OPERABILITY of the component. To establish OPERABILITY as used in this context means to either verify OPERABILITY as above or to return the required equipment to OPERABLE status and perform the associated surveillance requirements as required. Surveillances are specified to verify correct positioning of emergency chilled water system valves servicing equipment which is unit specific and common. It is not intended that surveillance of valves servicing equipment specific to the other Unit be required to establish ECWS OPERABILITY under this Technical Specification. The safety systems served by the emergency chilled water system cannot be considered OPERABLE in the absence of all (both normal and emergency) space cooling. The definition of OPERABILITY (TS 1.17) is to be used in conjunction with the Technical Specifications governing the affected systems, to determine the appropriate action requirement in the event of loss of all space cooling.

DESCRIPTION OF PROPOSED CHANGES NPF-10-130 AND NPF-15-130
AND SAFETY ANALYSIS

This is a request to delete License Condition 2.C(4) - Containment Tendon Surveillance.

Existing License Conditions

Unit 2

(4) Containment Tendon Surveillance (Section* 3.8.1, SER, SSER #5)

Within three (3) years of the date of issuance of this license. SCE shall (a) provide for NRC approval, and (b) implement, as appropriate, a tendon surveillance test program which will ensure full conformance with the provisions of Regulatory Guide 1.35 and Regulatory Guide 1.35.1. This tendon surveillance program shall include a specific program and commitments for re-tensioning of the tendons, such that the predicted prestressing force of each tendon will be greater than the required design prestressing force of each tendon for the entire plant life.

Unit 3

(4) Containment Tendon Surveillance (Section* 3.8.1, SER, SSER #5)

Within two (2) years of the date of issuance of this license, SCE shall provide for NRC approval, and within three (3) years of the date of issuance of this license, SCE shall implement a tendon surveillance test program which will ensure full conformance with the provisions of Regulatory Guide 1.35 and Regulatory Guide 1.35.1. This tendon surveillance program shall include a specific program and commitments for retensioning of the tendons, such that the predicted prestressing force of each tendon will be greater than the required design prestressing force of the tendon for the entire plant life.

DESCRIPTION

The existing tendon surveillance program for the San Onofre Nuclear Generating Station, Units 2 & 3 is consistent with the requirements of NRC Regulatory Guides 1.35 and 1.35.1. Specifically, the program conforms to these regulatory guides by incorporation of the following criteria:

a. If a surveillance tendon has a liftoff value below its lower tolerance band value, liftoff values for the two adjacent tendons will be measured.

b. If both liftoff values of the adjacent tendons are above their lower tolerance band value, the initial deviation will be considered acceptable.

c. If the initial liftoff value is lower than 90% of the lower tolerance band value, or the liftoff value of either of the adjacent tendons is below the lower tolerance band value, an investigation will be conducted.

d. The average prestress at all locations within the containment, as indicated by average liftoff values of tendons within an individual group of tendons, shall be maintained above the minimum design prestress.

License Condition 2.c(4) requires that SCE submit and implement a tendon surveillance program which will ensure full conformance with Regulatory Guides 1.35 and 1.35.1. This program, which has been briefly described in FSAR Section 3.8.1.7.2 and in Technical Specification 3/4.6.1.6, is described in detail in the report "Tendon Surveillance Requirements for San Onofre Nuclear Generating Station, Units 2&3" dated February, 1984. This report is hereby formally submitted (Attachment A). Also included is Reference 5 to Attachment A, "Experimental Determination of the Influence of Individual Tendon Stressing Upon Containment Post-Tensioning Strain, San Onofre Nuclear Generating Station, Units 2 and 3" dated February, 1984 (Attachment B). This formal submittal of the containment tendon surveillance program and implementation of this program, which began in January, 1982 at San Onofre Unit 2 and in February, 1983 at Unit 3, satisfy the intent of the license condition. Since this program satisfies all regulatory requirements, the need for re-tensioning of tendons, unless required by failure to satisfy program requirements during the life of the plant, is eliminated. All requirements of this license condition have been satisfied.

<u>Safety Analysis</u>

The proposed change discussed above shall be deemed to involve a significant hazards consideration if positive findings are made in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequence of an accident previously evaluated?

   <u>Response: No</u>

   License Condition 2.C(4) requires submittal and implementation of the San Onofre 2 & 3 tendon surveillance program. No physical change has been made to the physical plant. Elimination of the license condition as a result of satisfactory completion of its requirements in no way changes the commitments made in the tendon surveillance program or its compliance with regulatory requirements.

The containment prestressing system is described in FSAR Section 3.8.1.1.2. This prestressing system is provided so that the containment will withstand internal pressures resulting from postulated design basis primary or secondary energy releases inside containment (i.e. LOCA, MSLB, etc.). The tendon surveillance program ensures that the integrity of the tendons is maintained. Selected tendons are inspected and measurements of liftoff force are taken for comparison with predicted values considering expected long term stress losses. By ensuring that the measured liftoff values are within the prescribed tolerances, the average level of prestress is maintained above the minimum design value and well above the level actually required to ensure containment integrity. All deviations are investigated and reported in accordance with regulatory guide requirements as stated in approved station technical specifications.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change deletes a license condition which required submittal and implementation of the containment tendon surveillance program. The license condition has been satisfied by formal submittal of the program description, and the program has been implemented in compliance with Technical Specification 3/4.6.1.6. The license condition is thus unnecessary for long term plant operation. Its deletion in no way changes the tendon surveillance program or the physical plant from the configuration for which accident analyses were performed.

3. Will the operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety.

   Response: No

   As stated previously, deletion of this license condition in no way changes the tendon surveillance program or the physical plant, nor reduces the level of regulatory commitment governing its performance. The margin of safety for the containment prestressing system and its surveillance is thus unchanged.

48 FR 14870 dated April 6, 1983 provided examples of amendments not likely to involve a significant hazards consideration. This proposed change is considered to be most similar to example (1) in that it is an administrative change to delete License Condition 2.c(4) Containment Tendon Surveillance from the operating licenses for both San Onofre Unit 2 and Unit 3 because this license condition has been satisfied.

Safety and Significant Hazards Determination

Based on the Safety Analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Environmental Statement.

DMercurio:0907F

ATTACHMENT A

Tendon Surveillance Requirements

for the

San Onofre Nuclear Generating Station

Units 2&3

February 1984

(63 copies)

TENDON SURVEILLANCE REQUIREMENTS

FOR

SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 & 3

SOUTHERN CALIFORNIA EDISON

FEBRUARY, 1984

TABLE OF CONTENTS

I. INTRODUCTION

The SONGS 2 & 3 Containment Tendon Surveillance program was prepared consistent with requirements of NRC Regulatory Guides 1.35 Rev. 3 and 1.35.1 Rev. 1. This program was reviewed with the NRC in July, 1979. All methodologies and acceptance criteria of these regulatory guides were applied to determine individual stress losses and tolerance bands and to compare the prestress level which will exist during the 40 year expected plant life with that required by design. The comparison was made to demonstrate that lower bound prestress levels (i.e., assuming maximum value of predicted long-term stress loss plus tolerances) for the two primary groups of tendons would remain at levels equal to or greater than required by design.

During review of the technical specification in January, 1982, the NRC stated that their current position requires that the design prestress level must remain equal to or less than the predicted lower bound for each individual tendon curve rather than for the group average. In February, 1982, the NRC has required, as an operating license condition 2.C.(4) for San Onofre Unit 2, that:

> Within three (3) years of the date of issuance of this license, SCE shall (a) provide for NRC approval, and (b) implement, as appropriate, a tendon surveillance test program which will ensure full conformance with the provisions of Regulatory Guide 1.35 and Regulatory Guide 1.35.1. This tendon surveillance program shall include a specific program and commitments for re-tensioning of the tendons, such that the predicted prestressing force of each tendon will be greater than the required design prestressing force of the tendon for the entire plant life.

This report is provided in response to the NRC request to document compliance with the regulatory guides. It oulines the conservatism which is present in the existing system and its established surveillance program. The degree to which compliance with the new NRC requirement exists, and future required actions, are also addressed. In addition, a finite element analysis is described in which the total prestress was assumed to be reduced to correspond with the lowest calculated lower bound value for any of the surveillance tendons. Recommendations are provided concerning the acceptance criteria taking into account the finite element analysis results.

II. EXECUTIVE SUMMARY

The SONGS 2 & 3 Containment Prestressing System was designed using an average tendon force to determine the required number of tendons. Installation involved stressing each tendon to 80 percent of its ultimate capacity and releasing the tendon to seat against wedges which lock it into the anchor head. Average values of initial and long term stress losses were used to calculate average prestress levels which would exist throughout plant life. A surveillance program will be utilized to verify that actual losses are consistent with predicted values.

Individual prestress loss tolerance bands have been calculated for all surveillance tendons and average tolerance bands have been developed for all groups of tendons. A review of these curves demonstrated that the lowest margin existed for the horizontal (hoop) tendons in the cylinder section of the Unit 2 Containment Structure so a more detailed review was performed for these tendons. It was shown that the average predicted 40 year prestress level is more than 6 percent greater than that required by design. No individual tendon has a predicted prestress level in 40 years lower than the design prestress although some individual tendons may drop 2-5 percent below this value if actual losses that occur are at or near the maxiumum predicted values. Based on initial surveillance results for Unit 2 it would be expected that substantially less than 5 percent of the tendons may experience lift-off values in 40 years lower than required by design (if the actual losses are at or near maximum predicted values).

It has been shown that prestress existing at a particular point in the structure results from a stress field established by 34 or more tendons in the vicinity of that point, with no more than 6 percent contributed by any single tendon (Reference 5). Furthermore, a slight loss of prestress has only a negligible effect on the overall capacity of the structure and the liner plate strain level. The effect of all tendons being reduced in prestress to the lowest predicted value for any individual tendon was evaluated using the FINEL computer program. This analysis demonstrated that reinforcing steel stress and liner plate strain are well within allowable limits for this extreme condition.

Based on the above, it can be concluded that average prestress levels are an appropriate and conservative basis for assessing the adequacy of the Containment prestressing system. The degree to which average prestress levels are being maintained can be appropriately determined by monitoring the individual surveillance tendons against their individual prestress tolerance band curves. It can be further concluded that substantial design margin currently exists in the SONGS 2 & 3 prestressing system and that the planned surveillance program will effectively monitor the performance of that system. Retensioning of tendons is unnecessary, expensive, and such manipulations could potentially be harmful to this existing sytem.

III. DISCUSSION

A. System and Design Basis

Each of the SONGS 2 & 3 Containment Prestressing Systems consists of 90 inverted U-tendons and 114 hoop tendons (84 in the cylinder section or wall and 30 in the dome). Each tendon is made up of 55 strands with the VSL wedge anchoring system. The tendons are pre-installed by tensioning the tendon to a nominal value of 1820 kips which is equal to 80 percent of ultimate force. This initial tensioning establishes prestress levels throughout most of the length of the tendon. Internal prestress is somewhat lower than 1820 kips because of friction along the length of the tendon. Following tensioning the tendon is released to seat on wedges which lock the strands to the anchor head. In the vicinity of the anchor, some prestress loss occurs. Since friction is opposing relaxation at this point, equilibrium with the stress level achieved during stressing occurs a short distance from the anchorage.

The prestressing system was designed by VSL and the design calculation package was reviewed with the NRC Staff in March, 1979 and August, 1980, and the complete calculation package was provided for NRC Staff files in July, 1979. In this calculation, VSL provides a separate calculation for each of the three basic tendon groups -i.e., inverted U tendons, wall hoop tendons, and dome hoop tendons. These calculations assume an average value of stress loss which occurs during seating. This value is based on their overall experience with systems of this type. In fact, since precise control cannot be applied to the seating process, variations occur in the initial locked-in liftoff force which is measured immediately following anchorage. The calculations also assume an average value of sequence loss which is the loss resulting from elastic shortening which occurs when subsequent parallel tendons are tensioned. This loss is also not uniform since tendons which are tensioned first lose nearly 100 kips of prestress as a result of tensioning later tendons and the last tendon tensioned experiences no loss. The calculation also assumes losses as a result of long term creep, shrinkage and elastic shortening. Average values for these losses are also assumed, and, in this case, substantially lower variation is in fact experienced.

The final determination on number of tendons to be provided was based on these average calculated 40 year prestress values. An allowance of 4-7 percent was added for possible breakage of strands during stressing although < 0.1% strand breakage was actually experienced.

B. Installation and Final Prestress Levels

The tendons were installed in alternating groups such that in each location, no more than three tendons were installed at the same point in the sequence. Half of the U tendons were first tensioned followed by half the hoop tendons. The hoop tendons which had been passed over were tensioned next, followed by the remaining U tendons.

All tendons were tensioned to their specified maximum value and released to seat on the anchor head. Following seating, a lift off test was performed on each tendon to determine its locked-in force. While the target lift-off force was 1580 kips for the wall hoop tendons, for example, on Unit 2, these values ranged from 1505 kips to 1652 kips with an average of 1592 kips. Since the average was greater than required and the variation was randomly distributed, the level of prestress at all points within the structure is higher than required by design.

Using this lift-off force as a base, specific predicted tendon forces can be calculated for the full plant life. Specific values of sequence loss can be determined depending on the location of particular tendon in the stressing sequence. Long term losses can be calculated based on properties of materials used for construction. To account for variability, tolerances are used to establish maximum values of prestress loss (lower bound curve) and minimum loss values.

This has been done for each surveillance tendon with the predicted tendon force and upper and lower bounds plotted on semi-log paper. Two of these curves are attached to this report as figures 1 and 2. It is noted that these figures also indicate the design prestress level as determined from the VSL calculations taking into account the added tendons.

A review of these curves shows that in all cases, the predicted tendon lift-off force is greater than the lift-off corresponding to the design prestress for the full plant life. For the 20 surveillance tendons of Unit 2, 5 of the 40 lower bound values fall below the design prestress at some point during plant life. For Unit 3, only 3 lower bounds fall below design prestress. These tendons are all tendons which were among the earliest tensioned and have high sequence losses. Their initial lift-off forces are also average or below. Because of the stressing sequence, these tendons are surrounded by tendons which were tensioned later, and thus have low sequence losses and higher than average prestress. Thus the overall stress field in the vicinity of these tendons is at or above the average prestress level which served as a design basis for this system. Comparisons of average prestress levels for the tendon groups in Unit 2 are shown in figures 3, 4, and 5. In all cases the lower bound is at or above design prestress and the predicted level of prestress at 40 years is at least 6 percent above.

Measurements were also taken during installation of strain levels at several points in the structure. Although it is intuitively apparent that, for a large thick-walled concrete structure, strain levels at the location of a specific tendon cannot differ greatly from strain levels at adjacent and nearby tendons, these measurements provide proof of that point. Stressing of the tendon immediately adjacent to the strain gage produced only approximately 6 percent of the total strain ultimately experienced during prestressing operations. Furthermore it was noted that measurable contributions to the strain level were produced by stressing tendons 25 feet away from the gage. Thus, for the hoop tendons, about 34 adjacent tendons contribute to the strain at any one point in the structure and it is not possible for prestress level at a given point to be below the required average simply because a specific tendon at that point has a lower value. Although individual variations from the average are observed because of individual tendon histories, strain compatibility within the structure performs an automatic averaging function and provides justification for the use of average values both for design and system acceptance.

C. Surveillance Program and Acceptance Criteria

The SONGS 2 & 3 Tendon Surveillance program, which is based on and consistent with the NRC Regulatory Guides, is established to clearly detect an overall degradation in the prestressing system which would reduce prestress below the design level. It has been demonstrated

that prestress levels throughout the structure are effectively the average of all tendons in a given group. It has also been shown in figures 3, 4, and 5 that average prestress levels are predicted to be a least 6 percent greater than design prestress. Even on the extremely unlikely event that all tendons approach the lower bound, prestress levels will continue to be greater than required by the original design. It will, in fact, be demonstrated later in this report that a much lower level of design prestress can be used without exceeding allowable stress in reinforcing steel or, more importantly, allowable strain in the containment liner plate.

The program is established to detect abnormal degradation in the average prestress level by examining selected tendons at random for evidence of abnormal degradation on an inidividual basis. This is accomplished by comparing the lift-off force observed in that tendon with the value specifically predicted for that tendon. As long as these samples demonstrate that all lift-off forces are above the lower bound for each surveillance tendon, evidence is obtained that the overall system is being maintained above the lower bound and that adequate prestress exists.

It is important to distinguish this type of program from a random sampling program where several items of equivalent value, e.g. 3/4 inch Grade 60 reinforcing bars, are being sampled. In the latter case, since all bars should experience the same tensile force, a tensile force lower than expected is evidence of abnormality. It is not adequate in the case of the tendon surveillance to compare the lift-off force against an arbitrary design value. Each tendon has experienced a unique history and must be uniquely considered. A tendon predicted to have a lower bound value 15 percent greater than design prestress cannot be considered acceptable if its measured lift-off is 5 percent above design. In fact, such a value may be evidence of a serious system problem. Conversely, a tendon which evidences a lift-off force 2 percent below design prestress will be totally acceptable if its predicted lower bound was 4-5 percent below. The composite system is thus demonstrated to be functioning within its prescribed limits and with adequate margin.

The specific acceptance criteria of the SONGS 2 & 3 tendon surveillance program are outlined as follows:

1. Each tendon end lift-off force shall be above the lower bound. If a single lift-off force is below the lower bound but above 90% of the lower bound, two adjacent tendons shall be subjected to lift-off readings. If both of these adjacent lift-off forces are above the lower bound, the lift-off value for the initial tendon shall be considered a unique occurrence. This tendon shall be re-tensioned to the predicted value.

2. If a tendon lift-off reading is less than 90% of the lower bound value, it shall be de-tensioned and examined for excessive corrosion and/or wire/strand breakage. Following examination, it is re-tensioned to a value greater than the predicted prestress value for the corresponding surveillance year.

3. When two adjacent tendons are subjected to lift-off readings and either one of them shows a value below the lower bound, a comprehensive plan shall be developed to evaluate the complete pretensioning system.

4. Average lift-off forces for all surveillance tendons shall be above the design prestress value throughout the plant life. If the average falls below the design prestress value a comprehensive plan shall be developed to evaluate the complete pretensioning system.

Acceptance criteria 1) and 2) are intended to assure that there is no unusual deterioration of a single tendon. Acceptance criteria 3) and 4) are intended to assure that there exists sufficient prestress throughout the plant life and that no unusual deterioration of the overall prestressing system has taken place.

These criteria will effectively ensure the integrity of the containment structures for the following reasons:

1. If a tendon end force is within the established tolerance bands, it indicates that there is no unusual deterioration in that tendon.

2. If the predicted lower bound of a tendon falls below the design prestress level but is still above the lower bound, required prestress will still exist if the average lift-off value for all surveillance tendons is above the design prestress.

3. If a tendon end force is below the design prestress value, the membrane compression at that location is not necessarily below the design value. Only about 6% of the membrane compression at a point is due to the tendon at that point (Reference 5). In other words, if initial design provides a 6% margin in the prestress, complete loss of a tendon can be accommodated in the local area.

4. ACI code (Reference 6) states that actual losses have little effect on the strength of a member. Even if the average prestress force were below the design value, the strength of the containment will not be affected significantly, provided that no unusual deterioration (i.e., corrosion or wire/strand breakage) has taken place in the prestressing system.

5. The only effect reduced prestress will have is a slight increase in displacements of the shell liner strains, and reinforcement stresses under extreme/abnormal loading conditions.

Considering the above factors, the condition that a single tendon or even average prestress value is below the design prestress does not affect the structural integrity, and thus public safety, provided that stresses and strains in the other parts of the total system remain within the allowable limits.

IV. FINITE ELEMENT ANALYSIS

A finite element analysis was performed in order to demonstrate that a level of prestress that is less than the design prestress does not affect adversely the structural integrity of the containment pressure boundary. For this purpose the FINEL computer code was used. The computer program, the finite element model, and the method of application of loads was exactly the same as in the original analysis. The loading condition considered was the most critical loading combination for the shell and dome (i.e., the highest stresses occur under this loading) which is the Reference Loading Combination No. 10 of Reference (7).

Table 2 is a reproduction of the data already submmitted to the NRC (Reference 7). It is seen that the maximum reinforcement stress is 36 ksi under primary plus secondary loads. It is also seen that the concrete stresses and liner strains are not significant.

In order to perform a conservative analysis the overall prestress value was decreased to 94.5% of the design prestress value. This assumption implies that all tendon end forces will decrease to 1230 kips which is the lowest of lower bounds (Compared with the predicted tendon end forces ranging from 1311 kips to 1503 kips at the end of 40 year design life, Unit 2). Obviously, such an occurrence is extremely unlikely to happen but was assumed as a lower bound prestress.

Table 3 lists the results of this analysis. As can be seen from this table, the maximum reinforcement stress increases to 44.5 ksi under the primary plus secondary loads. Concrete stress and liner strains are still very low compared to their allowables.

V. SUMMARY AND CONCLUSIONS

A. The proposed acceptance criteria is a reasonable approach to this complicated problem.

B. If a tendon end force is within the established tolerance bands, it indicates that there is no unusual deterioration in that tendon. This can be most reliably detected by comparing the prestress level for a particular tendon with the predicted prestress level for that tendon.

C. If no unusual deterioration is noted, as evidenced by all surveillance tendons exhibiting lift-off values within their corresponding tolerance band, it can be assumed that the average stress level for the overall family of tendons represented by these surveillance tendons also remains within the average tolerance band. Since this average is always greater than the design prestress, adequate margin will exist within the structure.

D. If a tendon end force is below the design prestress value, the membrane compression at that location is not necessarily below the design value. This is because only about 6% of the membrane compression at a point is due to the tendon at that point (Reference 5). In other words, if initial design provides a 6% margin in the prestress, complete loss of a tendon can be accommodated in the local area. Sequence losses account for a high percentage of the calculated prestress loss. Since the tendons were tensioned in alternating groups those tensioned early, with higher prestress loss levels, are surrounded by tendons with much lower sequence losses. This assures that average stress levels throughout the structure are both relatively uniform and above design prestress. It has been demonstrated that local prestress is affected by tendons on either side of a particular location.

E. Allowing the predicted lower bound of a tendon end force to fall below the design prestress value does not in any way adversely impact the structural integrity of the containment pressure boundary.

F. Further analysis shows that, even with the assumption of all tendons having an end force equal to the lowest of lower bounds, the containment pressure boundary has ample design margin. Increase in reinforcement stress due to <u>one</u> tendon end force being at the lowest of lower bounds is within the accuracy of the calculations.

G. The requirement that all tendon end forces must be re-tensioned if they fall below the design value is overly conservative and would not significantly improve the integrity of the containment structure.

H. Re-tensioning more than 50 tendons per unit is viewed as an activity which could be potentially harmful to currently satisfactory containment structures. Risk of strand breakage or hardware damage exists during this operation. Strand breakage would reduce the current structural capacity and hardware damage or excessive strand breakage could require complete replacement of one or more tendons. Such manipulation of an existing system should only be undertaken if serious evidence of system degradation should be discovered.

VI. REFERENCES

1. Proposed Revision 3 to Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments", U.S.N.R.C. April, 1979.

2. Proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces For Inspection of Prestressed Concrete Containments", U.S.N.R.C. April, 1979

3. "Containment Structure In-Service Tendon Surveillance, San Onofre Nuclear Generating Station, Units 2 and 3", Becthel Spec. No. SO23-210-11 (SCE Spec. No. 59-9541) June 19, 1981.

4. Final Safety Analysis Report, San Onofre Nuclear Generating Station Units 2 and 3, Vol. 24, 16.4.6.1.6.1.

5. Experimental Determination of the Influence of Individual Tendon Stressing Upon Containment Post Tensioniong Strain. Bechtel National, Incorporated, February, 1984.

6. Commentary to Building Code Requirement For Reinforeced Concrete (ACI 318-77)

7. "Response to NRC Question 131.7, Table 131.7-2" Response to NRC Questions, San Onofre Nuclear Generating Station, Units 2 and 3, Vol. 2, Amendment 5.

VII. APPENDICES

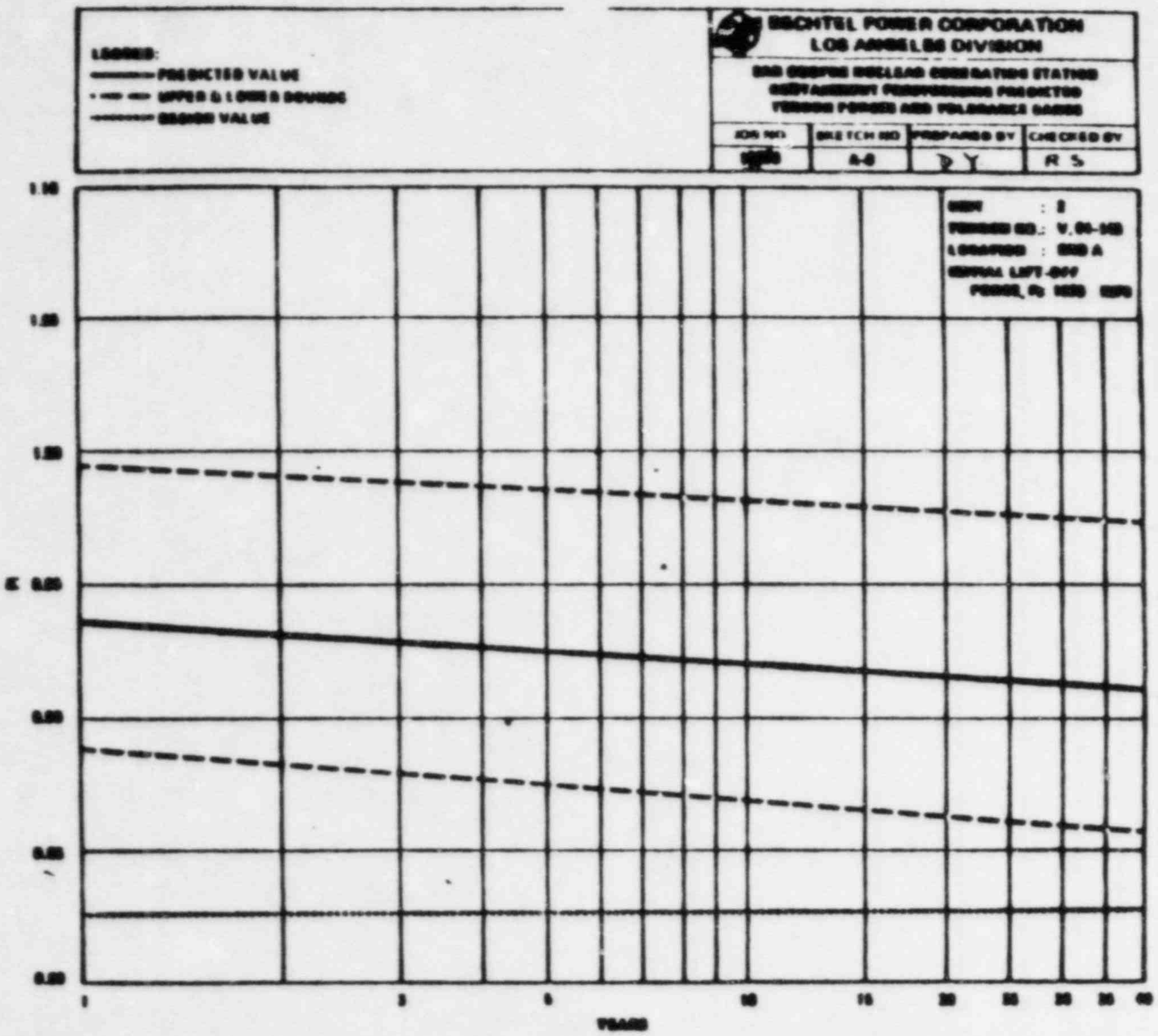


Fig. 1   Predicted and Design Prestress for a Vertical Tendon, Unit 2.

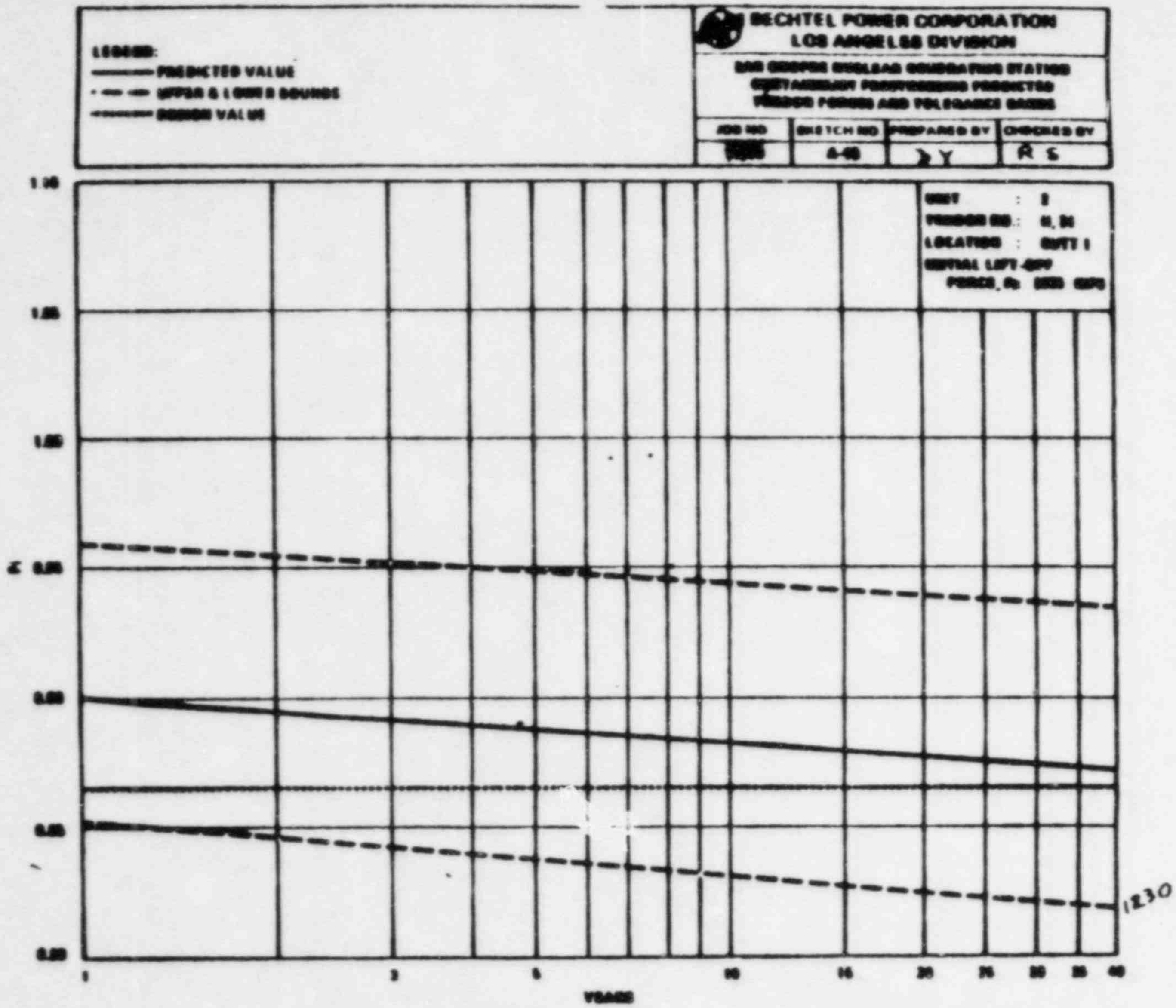


Fig. 2 Predicted and Design Prestress for a Horitonzal Tendon, Unit 2.

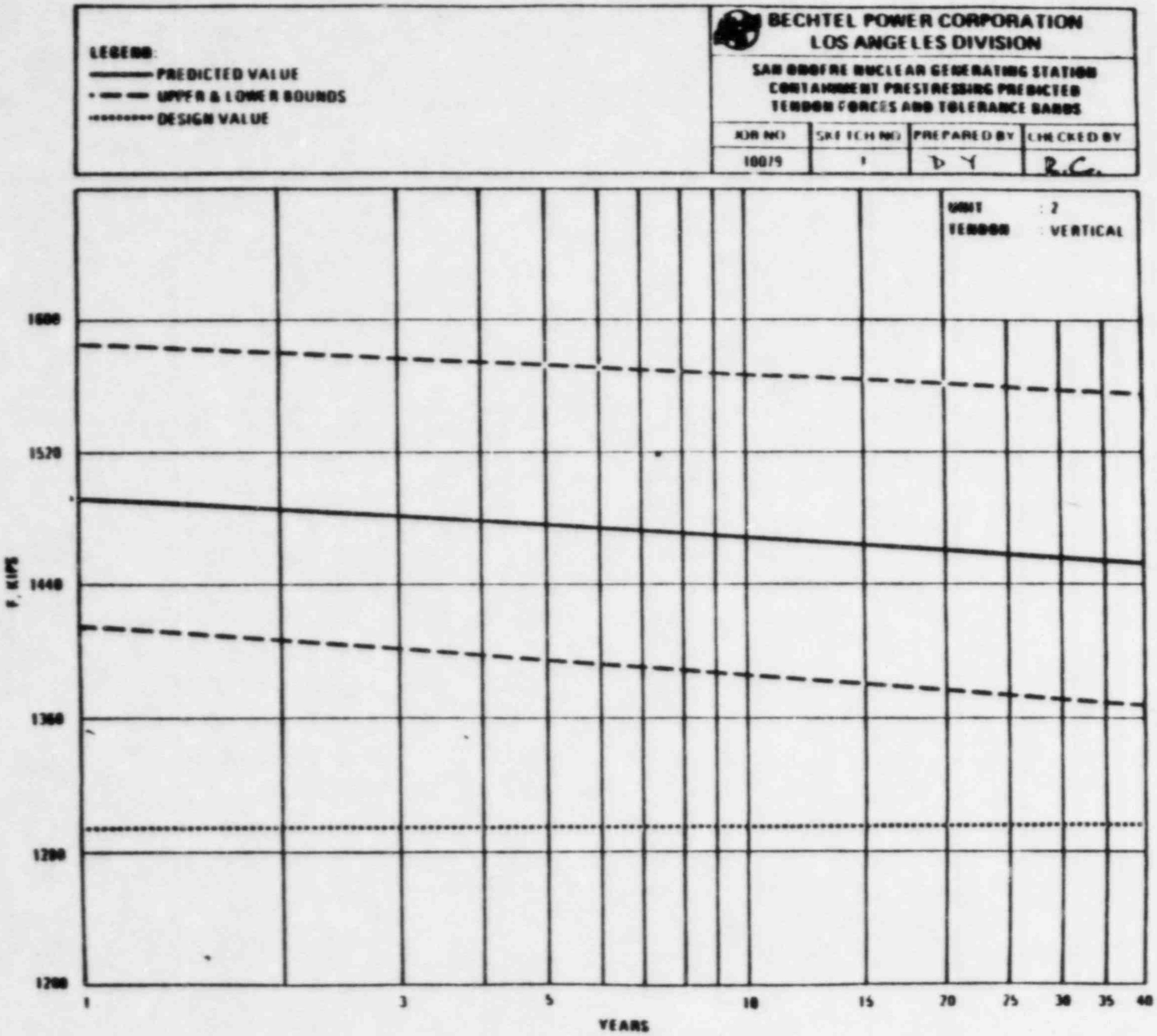


Figure 3 AVERAGE TOLERANCE BANDS OF 90 VERTICAL TENDONS, UNIT 2

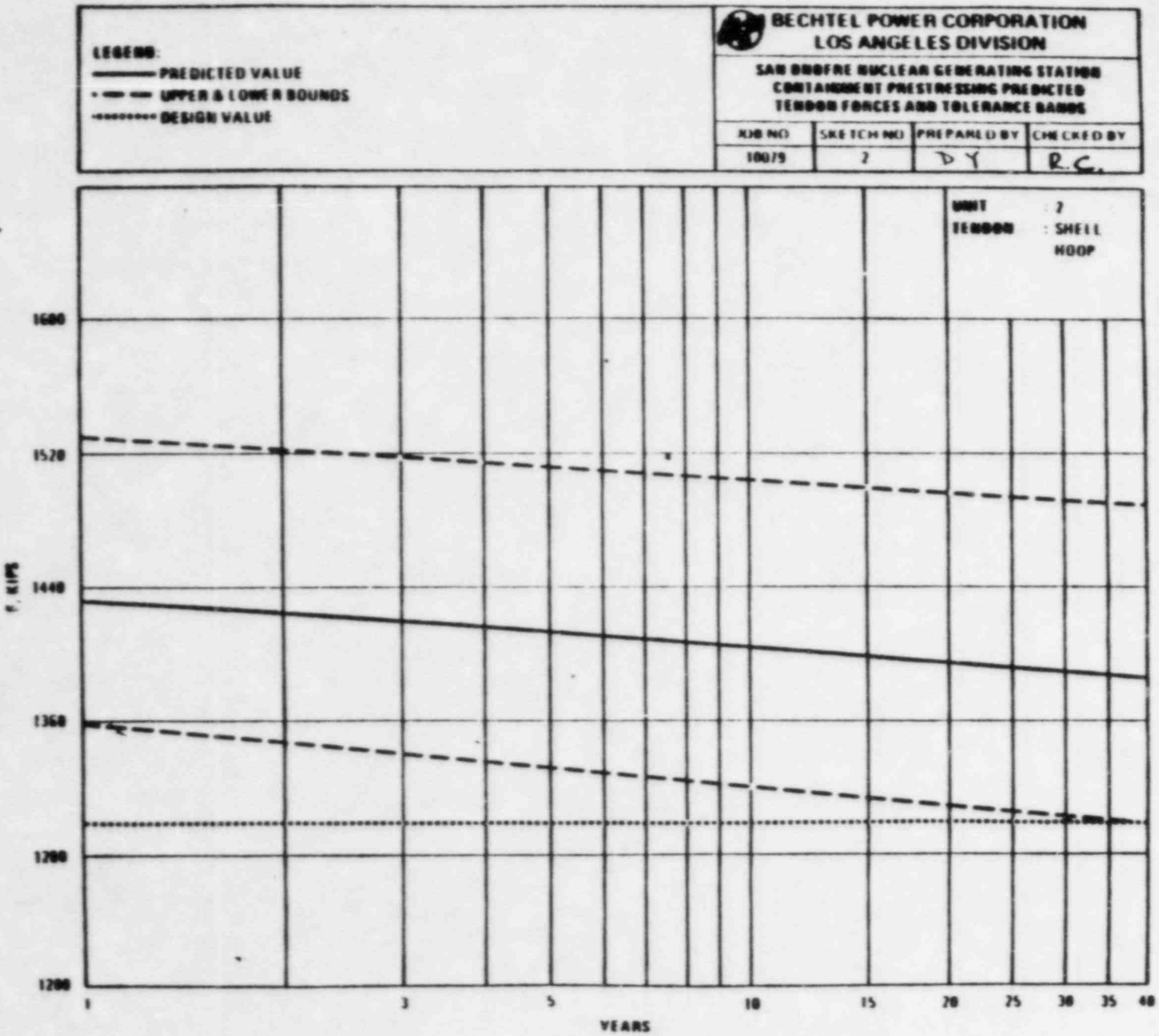


Figure 4 AVERAGE TOLERANCE BANDS OF 84 SHELL HOOP TENDONS, UNIT 2

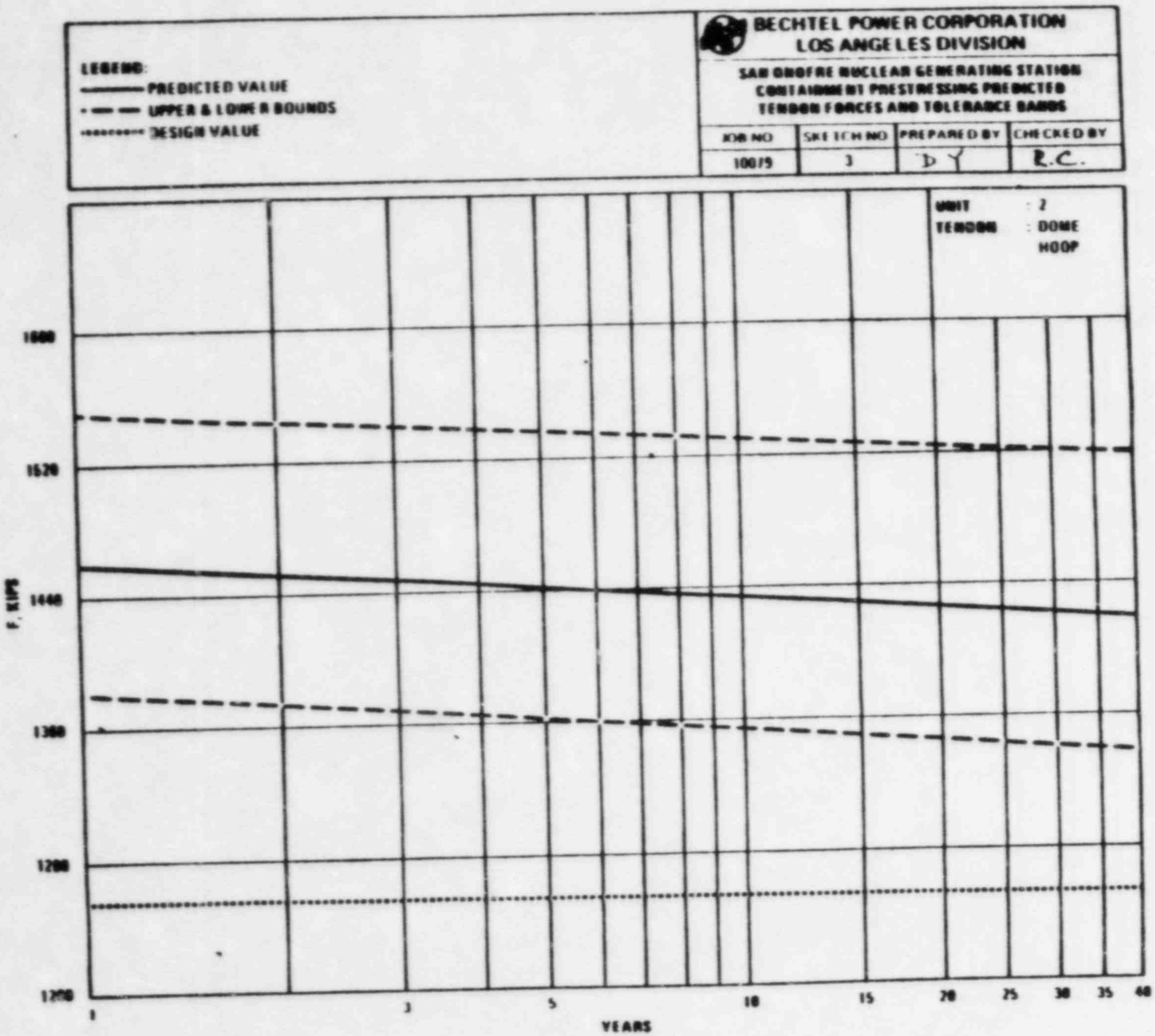


Figure 5 AVERAGE TOLERANCE BANDS OF 30 DOME HOOP TENDONS, UNIT 2

TABLE 1

TENDON SURVEILLANCE

		TENDON NUMBERS																			
	Years After Initial	1		3		5		10		15		20		25		30		35		40	
	Structural Integrity Test	H	V	H	V	H	V	H	V	H	V	H	V	H	V	H	V	H	V	H	V
		30	31-121	5	13-170	42	64-170														
		86	9-143	36	35-117	86	9-143	30	66-176	50	12-140	75	86-156	12	20-130	86	9-143	81	41-111	97	9-143
	Visual Inspection	97	66-176	79	6-50	75	94-140	86	9-143	114	5-57	86	9-143	90	70-172	31	9-170	109	90-152	86	31-121
		53	88-154	113	70-164	9	19-133	53	39-133	13	96-146	9	43-109	25	76-164	64	90-140	31	50-102	100	86-156
		64		87		100															
UNIT 2		30	31-121			42	64-170														
	Prestress	86	9-143			86	9-143	30	66-176			75	86-156			86	9-143			97	9-143
	Monitoring	97	66-176			75	94-140	86	9-143			86	9-143			31	64-170			86	31-121
	Tests	53	88-156			9	19-133	53	39-113			9	43-109			64	94-140			100	86-156
		64				100															
	Detensioning and Material Tests	97	88-154			42	19-133	30	66-176-			75	43-109			31	64-170			86	9-143
		53	66-176	7	23-130	31	19-133														
		64	88-154	30	47-105	64	88-154	69	11-141	42	95-147	90	25-127	100	39-113	80	43-190	64	88-154	6	51-101
	Visual Inspection	9	9-143	80	69-173	100	31-121	111	7-55	64	88-154	87	71-171	64	88-154	106	91-151	42	31-121	88	97-141
		97	39-113	94	83-159	75	65-177	2	76-166	97	43-109	4	31-121	9	86-156	24	19-133	86	9-143	32	50-102
		86		85		20															
UNIT 3		53	66-176			31	19-133														
	Prestress	64	88-154			64	88-154			42	95-147			100	39-113			64	88-154		
	Monitoring	9	9-143			100	31-121			64	88-154			64	88-154			42	31-121		
	Tests	97	39-113			75	65-177			97	43-109			9	86-156			86	9-143		
		86				20															
	Detension and Material Tests	53	66-176			31	19-133			97	95-147			9	39-113			64	88-154		

Table 131.7-2
ASME CODE, SECTION III, DIVISION 2,
STRESS ANALYSIS RESULTS (Sheet 6 of 8)
Reference Loading Combination No. 10[(b)]

		Concrete Stresses								Reinforcement Stresses								Liner Strains	
		Meridional				Hoop				Meridional				Hoop					
		Primary		Primary and Secondary		Primary		Primary and Secondary		Primary		Primary and Secondary		Primary		Primary and Secondary		Meri-dional $\times 10^{-6}$ in./in.	Hoop $\times 10^{-6}$ in./in.
...n	Section	Mem lb/in.^{2}	Mem & Ben lb/in.^{2}	Mem lb/in.^{2}	Mem & Ben lb/in.^{2}	Mem lb/in.^{2}	Mem & Ben lb/in.^{2}	Mem lb/in.^{2}	Mem & Ben lb./in.^{2}	Inside ksi	Outside ksi	Inside ksi	Outside ksi	Inside ksi	Outside ksi	Inside ksi	Outside ksi		
...ble		-3600	-4500	-4500	-5100	-3600	-4500	-4500	-5100	±54	±54	±54	±54	±54	±54	±54	±54	+2000 -4000	+2000 -4000
	1	89	(*)	102	(*)	100	(*)	109	(*)	11.4	18.2	11.4	18.2	12.5	7.7	11.8	17.5	290	330
	2	-179	-203	-174	-656	-66	500	-104	-511	-1.5	-1.1	-2.9	0.9	3.1	3.7	0.6	12.5	-310	-300
	3	-136	366	-136	-660	96	(*)	120	(*)	0.9	-1.5	-1.8	7.6	11.3	11.2	11.5	22.0	-250	90
	4	-157	209	-157	-758	160	(*)	149	(*)	-0.1	-2.1	-3.5	0.8	18.0	17.5	16.7	26.1	-200	290
	5	-212	51	-212	-1100	208	(*)	234	(*)	0.1	5.8	-3.7	14.6	27.2	26.1	25.0	36.0	-320	300
	6	-247	614	-247	-453	111	464	-266	-880	2.6	-8.0	-1.7	-3.5	2.1	1.9	-4.3	2.8	-200	-380
	7	-177	-603	-172	-800	67	141	30	-296	6.7	-6.4	4.8	-5.6	0.9	1.6	-1.1	2.8	370	30
	8	14	-435	-87	-465	61	6	17	-124	23.3	-2.7	16.3	-2.4	0.1	4.0	-1.7	7.4	590	50
Slab	9	-60	-787	-210	-1410	-23	-485	-98	-861	-3.9	9.6	-8.6	11.9	-1.5	4.1	-4.1	7.2	-540	-340
	10	-30	-945	-191	-1050	-32	-236	-80	-531	-2.7	7.4	-7.4	9.2	-0.8	1.7	-3.3	4.5	-370	-240
	11	-241	-303	-135	-203	4	(*)	73	(*)	-1.2	-3.0	-0.8	-1.2	0.7	0.6	4.2	5.3	20	140
...r ...y	12	102	(*)	72	(*)	100	(*)	74	(*)	1.1	1.4	3.6	10.8	1.4	1.5	3.9	10.8	80	90

...he increase in prestress forces due to elongation of tendon under 1.5 P_a internal pressure is not taken into account.

Table 131.7-2
ASME CODE, SECTION III, DIVISION 2,
STRESS ANALYSIS RESULTS (Sheet 6 of 8)
Reference Loading Combination No. 10[(b)]

		Concrete Stresses							
		Meridional				Hoop			
		Primary		Primary and Secondary		Primary		Primary and Secondary	
Portion	Section	Mem lb/in.^{2}	Mem & Ben lb/in.^{2}	Mem lb/in.^{2}	Mem & Ben lb/in.^{2}	Mem lb/in.^{2}	Mem & Ben lb/in.^{2}	Mem lb/in.^{2}	Mem & Ben lb./in.^{2}
Allowable		-3600	-4500	-4500	-5100	-3600	-4500	-4500	-5100
Dome	1	89	(*)	102	(*)	100	(*)	109	(*)
	2	-179	-203	-174	-656	-66	500	-104	-511
	3	-136	366	-136	-660	96	(*)	120	(*)
Wall	4	-157	209	-157	-750	160	(*)	149	(*)
	5	-212	51	-212	-1100	208	(*)	234	(*)
	6	-247	614	-247	-453	111	464	-266	-880
	7	-177	-603	-172	-800	67	141	30	-296
Base Slab	8	14	-435	-87	-465	61	6	17	-124
	9	-60	-787	-210	-1410	-23	-485	-98	-841
	10	-30	-945	-191	-1050	-38	-236	-80	-531
Reactor Cavity	11	-241	-303	-135	-303	4	(*)	73	(*)
	12	102	(*)	72	(*)	100	(*)	74	(*)

b. The increase in prestress forces due to elongation of tendon under 1.5 P_a internal pressure is not taken into account.

Table 131.7-2
ASME CODE, SECTION III, DIVISION 2,
STRESS ANALYSIS RESULTS (Sheet 6 of 8)
Reference Loading Combination No. 10[b]

Portion	Section	Reinforcement Stresses: Meridional, Primary, Inside ksi	Meridional, Primary, Outside ksi	Meridional, Primary and Secondary, Inside ksi	Meridional, Primary and Secondary, Outside ksi	Hoop, Primary, Inside ksi	Hoop, Primary, Outside ksi	Hoop, Primary and Secondary, Inside ksi	Hoop, Primary and Secondary, Outside ksi	Liner Strains: Meridional $X10^{-6}$ in./in.	Liner Strains: Hoop $X10^{-6}$ in./in.
Allowable		±54	±54	±54	±54	±54	±54	±54	±54	+2000 -4000	+2000 -4000
Dome	1	11.4	10.2	11.4	10.2	12.5	7.7	11.8	17.5	290	330
	2	-1.5	-1.1	-2.9	0.9	3.1	3.7	0.6	12.5	-310	-300
	3	0.9	-1.5	-1.8	7.6	11.3	11.2	11.5	22.0	-250	90
Wall	4	-0.1	-2.1	-3.5	0.8	18.0	17.5	16.7	26.1	-200	290
	5	0.1	5.8	-3.7	14.6	27.2	26.1	25.0	36.0	-320	300
	6	2.6	-8.0	-1.7	-3.5	2.1	1.9	-4.3	2.8	-280	-380
	7	6.7	-6.4	4.8	-5.6	0.9	1.6	-1.1	2.8	370	30
Base Slab	8	23.3	-2.7	[illegible]	-2.4	0.1	4.0	-1.7	7.4	590	50
	9	-3.9	9.6	-8.6	11.9	-1.5	4.1	-4.1	7.2	-540	-340
	10	-2.7	7.4	-7.4	9.2	-0.8	1.7	-3.3	4.5	-370	-240
Reactor Cavity	11	-1.2	-3.0	-0.8	-1.2	0.7	0.6	4.2	5.3	20	140
	12	1.1	1.4	3.6	10.8	1.4	1.5	3.9	10.8	80	90

b. The increase in tendon under 1.5 P_a internal pressure is not taken into account.

ASME CODE, SECTION III, DIVISION 2, STRESS ANALYSIS RESULTS

REFERENCE LOADING COMBINATION[(1)]

$D + L + F^{(2)} + 1.5P_A + T_A + R_A$

PORTION	SECTION	CONCRETE STRESSES								REINFORCEMENT STRESSES								LINER STRAINS	
		MERIDIONAL				HOOP				MERIDIONAL				HOOP				MERI-DIONAL	HOOP
		PRIMARY		PRIMARY & SECONDARY		PRIMARY		PRIMARY & SECONDARY		PRIMARY		PRIMARY & SECONDARY		PRIMARY		PRIMARY & SECONDARY			
		Mem psi	Mem + Ben psi	Mem psi	Mem + Ben psi	Mem psi	Mem + Ben psi	Mem psi	Mem + Ben psi	In-side ksi	Out-side ksi	In-side ksi	Out-side ksi	In-side ksi	Out-side ksi	In-side ksi	Out-side ksi	$X10^{-6}$ in/in	$X10^{-6}$ in/in
ALLOWABLE		-3600	-4500	-4500	-5100	-3600	-4500	-4500	-5100	±54	±54	±54	±54	±54	±54	±54	±54	+2000 -4000	+2000 -4000
DOME	1	128	(*)	139	(*)	140	(*)	148	(*)	15.0	11.2	14.5	21.3	15.9	10.8	15.4	21.0	444	476
	2	-130	-42	-122	-437	31	716	-45	-244	0.1	-1.1	-1.1	0.6	4.4	5.0	2.2	14.2	-130	-240
	3	-92	438	-90	-527	149	(*)	168	(*)	1.3	-1.8	-1.2	7.5	15.5	14.9	15.6	26.0	-167	256
WALL	4	-111	279	-111	-661	223	(*)	213	(*)	0.7	-1.7	-2.3	0.7	23.7	23.0	22.9	32.2	-135	520
	5	-165	67	-165	-1030	209	(*)	229	(*)	1.5	7.1	-2.2	16.7	34.9	33.5	33.9	44.5	-237	547
	6	-200	-293	-200	-574	197	539	-154	-674	3.1	-0.4	-1.0	-4.3	3.0	2.9	-3.3	3.9	-149	-300
	7	-151	-1189	-146	-989	37	132	8	-275	9.4	-7.5	7.4	-6.7	1.0	1.6	-0.9	2.9	556	84
BASE SLAB	8	20	-483	-82	-400	59	10	19	-111	24.0	-3.1	17.3	-2.6	0	4.1	-1.7	7.5	614	47
	9	-55	-821	-207	-1440	-26	258	-100	-870	-4.1	9.5	-0.7	11.4	-1.6	4.2	-4.3	7.3	-554	-352
	10	-32	-498	-196	-1070	-28	-247	-89	-549	-2.9	6.1	-7.5	9.4	-0.8	1.3	-3.4	4.4	-376	-243
REACTOR	11	-245	-273	-159	-235	5	(*)	71	(*)	-1.1	-1.7	-0.8	-1.4	0.7	0.7	4.1	5.3	9	132
CAVITY	12	98	(*)	82	(*)	99	(*)	86	(*)	1.7	-0.7	4.4	11.5	1.7	-0.6	4.6	11.4	110	127

(1) The increase in prestress forces due to elongation of tendon under $1.5P_A$ internal pressure is not taken into account.
(2) F = (0.9454)(Final Prestress)
* Sections are fully cracked.

ATTACHMENT B

Experimental Determination of the Influence of Individual

Tendon Stressing Upon Containment Post-Tensioning Strain

San Onofre Nuclear Generating Station

Units 2&3

February 1984

(63 copies)

DESCRIPTION OF PROPOSED CHANGE NPF-10-134 AND NPF-15-134
AND SAFETY ANALYSIS

This is a request to revise Technical Specification 2.2.2, Table 2.2-2, Core Protection Calculator Addressable Constants.

Existing Specifications

Units 2 and 3

See Attachment A. The existing specification is identical for both units.

Proposed Specifications

Units 2 and 3

See Attachment B. The proposed specification is identical for both units.

Description

As a part of the design and operation of the Core Protection Calculator (CPC) system a set of operational constants are generated to insure safe and proper operation of the system. A limited number of these constants are designated as "addressable" constants because of the high probability or necessity for changes to these constants during startup and normal fuel cycle operations. The CPC addressable constants are provided to allow calibration of thermal and neutron flux powers and Reactor Coolant System (RCS) flowrate to updated, measured values; to allow adjustment of CEA shadowing factors, radial peaking factors, and power distribution synthesizing coefficients based on test measurement results; to allow adjustment of DNBR and Local Power Density (LPD) pre-trip setpoint based on plant monitoring equipment operations; and to allow adjustment to power penalty factors and control element assembly (CEA) deviation penalties based on plant operating conditions and plant equipment status.

The purpose of the proposed technical specification change is to allow changes to the value of the addressable flow calibration constant, FC2, without approval of the Onsite Review Committee (OSRC). The value of FC2 is currently specified as 0.0. This change requests that FC2 be indicated as a range ($\leq$ 0.0) consistent with the intent of this table for defining allowable ranges of adjustment not requiring OSRC approval. Evaluation of the coolant flow verification tests performed during startup testing has indicated that the CPC's are calculating higher than measured flow rates during flow coast downs. Expanding the allowable range for the flow calibration bias would allow implementation of a small set of values for FC2 without having to convene the OSRC. This change enables the CPC calculated flow to be adjusted more conservatively consistent with the original intent.

Safety Evaluation

The proposed changes discussed above shall be deemed to involve a significant hazards consideration if positive findings are made in any of the following areas:

1. Will operation of the facility in accordance with these proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The new allowable range for FC2 results in a reduction in the CPC calculated flow. Adjustment of FC2 in conjunction with any applicable SONGS CPC calibration procedures moves the calculated DNBR value closer to the fixed DNBR trip setpoint, thus providing a more conservative DNBR margin to trip for operating conditions. Therefore, the proposed change will not increase the probability of occurrence or the consequences of an accident/malfuncion as previously stated in Chapter 15 of the SONGS FSAR.

2. Will operation of the facility in accordance with these proposed changes create the possiblity of a new or different kind of accident from any accident previously evaluated?

   Response: No

   These proposed changes specify CPC addressable constants because the CPC responds to design basis events, these changes do not create the possiblilty for a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with these proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The Reactor Coolant Flow Algorithm of the Core Protection Calculator (CPC) calculates a normalized mass flow rate based on reactor coolant pump (RCP) speeds and primary coolant specific volume for each leg of the reactor coolant system and for the reactor core. In addition, the algorithm calculates the time derivative of core flow rate to determine a projected value of Departure from Nucleate Boiling Ratio (DNBR). If this DNBR value is below a predetermined setpoint, the CPC initiates a reactor trip signal.

To verify the proper operation of the Reactor Coolant Flow Algorithm, hot functional loss of flow testing was performed in which calculated normalized core mass flow rates from each of the four CPC channels are compared with measured normalized flow rates determined from differential RCP pressures and

RCP speeds. If the CPC's compute flow to be higher than what is actually measured at any time during the coastdown (i.e., if the calculated flow is nonconservative with respect to measured flow), the current Startup Test Requirements state that the CPC addressable calibration coefficients, FC_1 and FC_2, must be adjusted to ensure that the CPC values of flow are less than or equal to the measured values. These requirements were established to meet set guidelines. FC_1 is a gain term that models the slope of calculated normalized flow rates versus measured normalized flow rates; FC_2 is an offset which, when set less than zero, forces the CPC to compute lower flow rates.

Analysis of the hot functional flow coastdown test results at SONGS produced the following interim values for the CPC flow calibration addressable constants (these values were later adjusted based on steady state flow measurements):

	Unit 2	Unit 3
FC_1	1.0	1.0
FC_2	-0.015	-0.020

These values ensure conservative CPC flow rate calculations during coastdowns as required by the Acceptance Criteria of the Startup Test Requirements. Therefore, the CPC will provide a greater degree of protection for the FSAR design basis events and the margin of safety provided by the CPC is increased.

The proposed change to the value of the addressable flow calibration constant, FC2, is similar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it is a change resulting from a small refinement of a previously used calculation model.

Safety and Signifiant Hazards Determination

Based on the Safety Evaluation, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Environmental Statement.

HRP:1138F:2879u

ATTACHMENT A

(Existing Specification)

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	≤ 1.15
61	FC2	Core coolant mass flow rate calibration constant	0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	≥ 1.02
64	TPC	Thermal power calibration constant	≥ 0.90
65	KCAL	Neutron flux power calibration constant	≥ 0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted

ATTACHMENT B

(Proposed Specification)

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	≤ 1.15
61	FC2	Core coolant mass flow rate calibration constant	$\triangleq 0.0$
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	≥ 1.02
64	TPC	Thermal power calibration constant	≥ 0.90
65	KCAL	Neutron flux power calibration constant	≥ 0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted

DESCRIPTION OF PROPOSED CHANGE NPF-10-135 AND NPF-15-135
AND SAFETY ANALYSIS

This is a request to revise Technical Specification Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Values.

Existing Specifications

Units 2 and 3

See Attachment A. The existing specification is identical for both units.

Proposed Specifications

Units 2 and 3

See Attachment B. The proposed specification is identical for both units.

Description

This change modifies the setpoints for the Control Room Toxic Gas Isolation System (TGIS). Setpoints are given in Technical Specification 3/4.3.2, Table 3.3-4, Engineered Safety Feature Actuation System Instrumentation.

The change is the result of a desire to increase setpoints to facilitate equipment calibration. In the original evaluation, no parametric analysis was performed to determine the highest detector setpoints which could be selected to meet the 2 minute toxicity limit in the control room. The parametric analysis has now been performed. In addition, a more realistic flow model has been used to analyze the control room leakage.

Since most of the control room boundary is not directly exposed to the outside, a model was developed which takes credit for the dilution which occurs as air flows from the outside through the areas adjacent to the control room boundary. The leak sources, leak rates and dilution volumes are described in FSAR Table 6.4-1. The model was previously discussed with the NRC.

The extent to which the analyzer setpoints could be raised and still provide adequate protection to the control room operators was evaluated. The acceptance criterion for the analysis was that for the two minute period following the time when the analyzer actuates an alarm, the control room concentration should not exceed the two minute toxicity limit for each evaluated chemical. In the case of chlorine, aqueous ammonia and hydrocarbons, toxic gas concentration buildup in the control room lags the monitor response. Therefore, a setpoint in excess of the toxicity limit is justified assuming a rupture of the largest container described in the FSAR Table 6.4-3. For chlorine, a setpoint in excess of the 15 ppm toxicity limit

is justified based on the rupture of the largest tank, however, a continuous slow release could lead to a control room concentration just reaching the 15 ppm level. Therefore, the allowed value for chlorine has been limited to 15 ppm. Similarly for aqueous ammonia, a setpoint in excess of the toxicity limit is justified based on the largest spill. Again the allowed value is limited to the two minute toxicity limit of 100 ppm. For hydrocarbons the allowed value is restricted to 200 ppm which is below the toxicity limit (750 ppm) for propane and butane which were analyzed in FSAR Section 6.4. This provides an allowance for other hydrocarbons which may have lower setpoints.

For carbon dioxide, the analysis shows that even with no control room isolation, the maximum control room concentration is 11,000 ppm. Since the two minute toxicity limit for carbon dioxide is 50,000 ppm, this monitor has been deleted from the Technical Specifications.

Since the revised setpoints results in a control room concentration during the first two minutes after the detector response which is lower than the toxicity limit, adequate protection is provided for the control room operator. In addition, there is a lower probability of spurious detector actuation which would lead to a disruption of normal plant operation. This is viewed as a positive measure to minimize unwarranted distractions to plant personnel.

Safety Evaluation

The proposed changes discussed above shall be deemed to involve a significant hazards consideration if positive findings are made in any of the following areas:

1. Will operation of the facility in accordance with these proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The probability of occurrence of a toxic chemical release will be unaffected by the proposed change to the Technical Specifications since the function of the detectors is to mitigate the consequences of accidents rather than prevent an accident. The consequences of a postulated accident will not be increased over that previously analyzed since the new detector setpoints will still provide the operators with 2 minutes of warning prior to the time that the toxicity limit is reached inside the control room. This is sufficient time for the operators to don self-contained breathing apparatus.

2. Will operation of the facility in accordance with these proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed setpoint change will not result in the initiation of an accident.

3. Will operation of the facility in accordance with these proposed changes involve a significant reduction in a margin of safety?

Response: No

As discussed in the description, the revised setpoints still allow sufficient time for the operators to respond to an alarm indicating a potential hazard due to chemical release resulting from the rupture of the design basis containers or resulting from the slow release of chemicals due to small leaks. The control room concentration does not exceed the two minute toxicity limit for at least two minutes following the analyzers' response to the setpoint concentration of the chemicals. No significant reduction in the margin of safety results from revising the setpoints.

The proposed change to the TGIS setpoints is similar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it is a change resulting from a small refinement of a previously used calculation model.

Safety and Significant Hazards Determination

Based on the Safety Evaluation, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Environmental Statement.

HRP:1139F

ATTACHMENT A

(Existing Specification)

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
9. CONTROL ROOM ISOLATION (CRIS)		
a. Manual CRIS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Airborne Radiation		
i. Particulate/Iodine	$\leq 5.7 \times 10^4$ cpm**	$\leq 6.0 \times 10^4$ cpm**
ii. Gaseous	$\leq 3.8 \times 10^2$ cpm**	$\leq 4.0 \times 10^2$ cpm**
d. Automatic Actuation Logic	Not Applicable	Not Applicable
10. TOXIC GAS ISOLATION (TGIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Chlorine - High	$\leq$ 6.0 ppm	$\leq$ 6.2 ppm
c. Ammonia - High	$\leq$ 42.4 ppm	$\leq$ 44.7 ppm
d. Butane/Propane - High	$\leq$ 84.8 ppm	$\leq$ 89.3 ppm
e. Carbon Dioxide - High	$\leq$ 4061.3 ppm	$\leq$ 4275.0 ppm
f. Automatic Actuation Logic	Not Applicable	Not Applicable

ATTACHMENT B

(Proposed Specification)

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
9. CONTROL ROOM ISOLATION (CRIS)		
a. Manual CRIS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Airborne Radiation		
i. Particulate/Iodine	$\leq 5.7 \times 10^4$ cpm**	$\leq 6.0 \times 10^4$ cpm**
ii. Gaseous	$\leq 3.8 \times 10^2$ cpm**	$\leq 4.0 \times 10^2$ cpm**
d. Automatic Actuation Logic	Not Applicable	Not Applicable
10. TOXIC GAS ISOLATION (TGIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Chlorine - High	14.3 ≤ ~~[illegible]~~ ppm	15.0 ≤ ~~[illegible]~~ ppm
c. Ammonia - High	97 ≤ ~~42.4~~ ppm	100 ≤ ~~44.7~~ ppm
d. Butane/Propane - High	193 ≤ ~~84.8~~ ppm	200 ≤ ~~89.3~~ ppm
~~e. Carbon Dioxide - High~~ deleted	~~≤ 4061.2 ppm~~ deleted	~~≤ 4276.0 ppm~~ deleted
~~f.~~ e. Automatic Actuation Logic	Not Applicable	Not Applicable

DESCRIPTION OF PROPOSED CHANGE NPF-10-138 AND NPF-15-138
AND SAFETY ANALYSIS

This is a request to revise Technical Specifications 3.1.3.1, "CEA Position", 3.1.3.6, "Regulatory CEA Insertion Limits", and 3.1.3.7, "Part Length CEA Insertion Limits."

Existing Specifications

Unit 2

See Attachment A

Unit 3

See Attachment C

Proposed Specifications

Unit 2

See Attachment B

Unit 3

See Attachment D

Description

The purpose of these changes is to modify the Core Protection Calculators (CPC's) and Control Element Assembly Calculators (CEAC's) to reduce the sensitivity to electrical noise. These changes would substantially improve the availability of San Onofre Units 2 and 3 through the reduction of spurious trips and the resulting evaluation and restart time.

The CPC and Core Operating Limit Supervisory System (COLSS) are responsible for the safety and monitoring functions, respectively, of the reactor core. COLSS monitors the DNB Power Operating Limit (POL) and various operating parameters to help the operator maintain plant operation within the limiting conditions for operation (LCO). Operating within the LCO guarantees that in the the event of an Anticipated Operational Occurrence (AOO), the CPC's will provide a reactor trip in time to prevent unacceptable fuel damage.

The COLSS reserves the Required Overpower Margin (ROPM) to account for the Loss of Flow (LOF) transient which is the most limiting AOO for San Onofre. When the COLSS is Out Of Service (COOS) the monitoring function is performed via the CPC calculation of DNBR in conjunction with the Technical Specification COOS Limit Line which restricts the reactor power sufficiently to preserve the ROPM.

These proposed changes reduce the sensitivity of the CEAC's to inward deviations of the CEA's. This involves setting all the inward CEAC CEA deviation penalty factors to 1.0. An inward CEA deviation event would thus not be accompanied by the application of a CEA deviation penalty in the CPC DNB-OPM and Linear Heat Rate (LHR) calculations. The protection for an inward CEA deviation event is accounted for separately.

If an inward CEA deviation event occurs, the current CPC algorithm applies two penalty factors to the DNBR and LHR calculations. This first, a static penalty factor, is applied upon detection of the event. The second, a xenon redistribution penalty, is applied linearly as a function of time after the CEA drop. The expected margin degradation for the inward CEA deviation event resulting from this change is accounted for in two ways in the proposed change. The ROPM reserved in COLSS is used to account for some of the margin degradation. When the combination of the static and Xenon redistribution penalties exceed the reserved ROPM, a power reduction in accordance with the appropriate curve in Figure 3.1-1A is performed. In addition, the part length CEA maneuvering is restricted in accordance with Figure 3.1-3 to justify reduction of the Part Length Rod (PLR) deviation penalty factors.

Safety Evaluation

The proposed changes discussed above shall be deemed to involve a significant hazards consideration if positive findings are made in any of the following areas:

1. Will operation of the facility in accordance with these proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The reduction of the inward CEA deviation penalty factors is compensated for by the reservation of safety margin (i.e., ROPM) and by providing for a reduction in core power when the actual CEA deviation penalty exceeds the ROPM allowance. Hence, these proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with these proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed changes only affect the CEA deviation event and do not create the possibility of any new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with these proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The reduction of the inward CEA deviation penalty factors is compensated for by the reservation of safety margin (i.e., ROPM) and by providing for a reduction in core power when the actual CEA deviation penalty exceeds the ROPM allowance. FSAR Sections 7.2.1.1.2.2 "Control Element Assembly Position Measurements," 7.2.1.1.2.5 "Core Protection Calculators," and 7.2.1.1.2.6 "Software Design" provides a more complete discussion of CEA deviation penalty factors. FSAR Section 15.4.1.3 "Control Element Assembly Misoperation" provides a detailed account of accident sequences. Therefore, these proposed changes do not involve a significant reduction in a margin of safety.

48FR 14864 dated April 6, 1983 provided examples of amendments that are not likely to involve a significant hazards consideration. Although these changes do not increase the probability or consequences of any previously analyzed accident, they would most likely be considered to be most similar to example (vi) in that the reduction in technical specification requirements may be perceived to insignificantly reduce in some way a safety margin.

Safety and Significant Hazards Determination

Based on the above discussion, Proposed Changes NPF-10-138 and NPF-15-138 do not involve a significant hazards consideraton in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; 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. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

CEWilliams:0757F

ATTACHMENT A

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and regulating) CEAs, and all part length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full length or part length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one full length or part length CEA misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

   2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

   Otherwise, be in at least HOT STANDBY within 6 hours.

*See Special Test Exceptions 3.10.2 and 3.10.4.

ACTION: (Continued)

d. With one or more full length or part length CEAs misaligned from any other CEAs in its group by more than 7 inches but less than or equal to 19 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

e. With one full length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

f. With one full length CEA inoperable due to causes other than addressed by ACTION a. above, but within its above specified alignment requirements and either withdrawn to greater than or equal to 145 inches or within the Long Term Steady State Insertion Limits if in full length CEA group 6, operation in MODES 1 and 2 may continue.

g. With one part length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 7 inches (indicated position) of all other part length CEAs in its group.

REACTIVITY CONTROL SYSTEMS

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2, with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. Less than or equal to 14 Effective Full Power Days per calendar year.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

   1. Restore the regulating CEA groups to within the limits, or

   2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using the above figure.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

---

*See Special Test Exceptions 3.10.2 and 3.10.4.

#With K_{eff} greater than or equal to 1.0.

ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per calendar year, either:

   1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

   2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The position of the part length CEA group shall be:

a. withdrawn to $\geq$ 145" or;

b. restricted to prevent the neutron absorber section of the part length CEA group from covering the same axial segment ($\leq$ 145") of the fuel assemblies for a period in excess of 7 EFPD out of any 30 EFPD period.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the neutron absorber section of the part length CEA group covering any same axial segment of the fuel assemblies for a period exceeding 7 EFPD out of any 30 EFPD period, either:

a. Reposition the part length CEA group to ensure no neutron absorber section of the part length CEA group is covering the same axial segment of the fuel assemblies within 2 hours, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA group shall be determined at least once per 12 hours.

ATTACHMENT B

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and regulating) CEAs, and all part length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full length or part length CEA inoperable or misaligned from an other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one full length or part length CEA misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with Figure 3.1-1A and that within one hour the misaligned CEA is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

   2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a. Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

      b. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

   Otherwise, be in at least HOT STANDBY within 6 hours.

* See Special Test Exceptions 3.10.2 and 3.10.4.

ACTION: (Continued)

d. With one or more full length or part length CEAs misaligned from any other CEAs in its group by more than 7 inches but less than or equal to 19 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with Figure 3.1-1A and that within one hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

a. Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

b. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

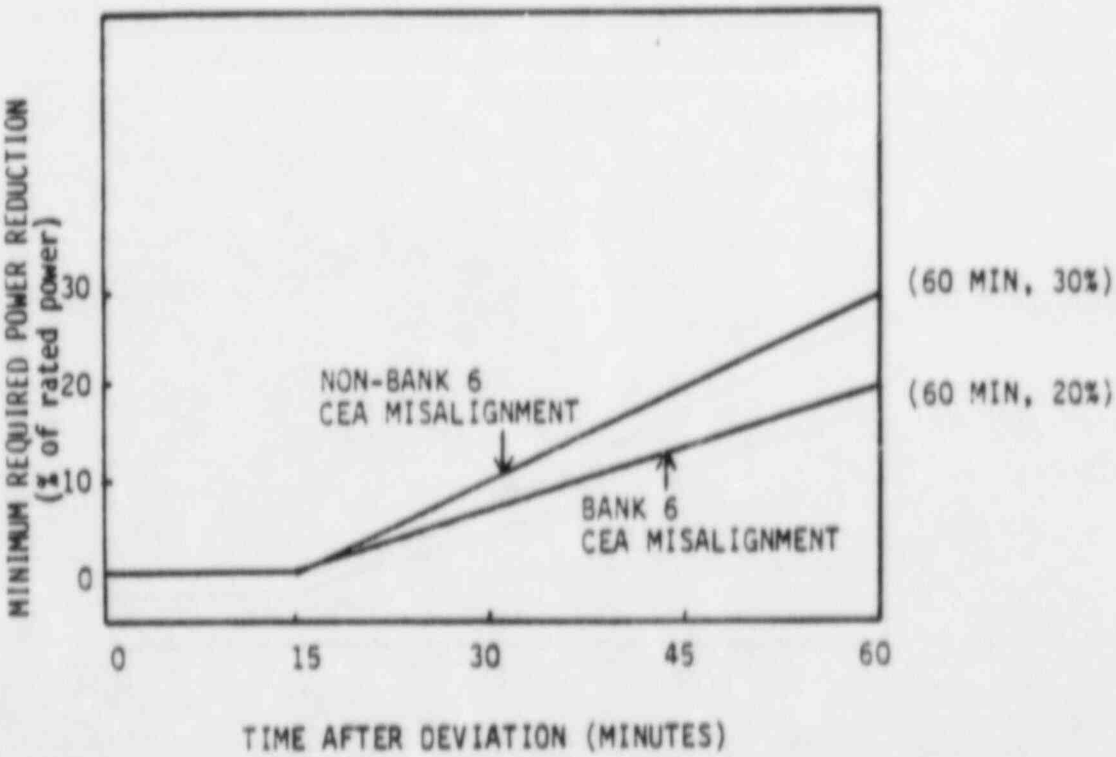
e. With one full length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

f. With one full length CEA inoperable due to causes other than addressed by ACTION a. above, but within its above specified alignment requirements and either withdrawn to greater than or equal to 145 inches or within the Long Term State Insertion Limits if in full length CEA group 6, operation in MODES 1 and 2 may continue.

g. With one part length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 7 inches (indicated position) of all other part length CEAs in its group.

FIGURE 3.1 - 1A

Required Power Reduction after CEA Deviation*



* When core power is reduced to 60% of rated power per this limit curve, further reduction is not required by this specification.

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2, when COLSS is in service and to the Short Term Steady State Insertion Limits when COLSS is out-of-service with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day Interval, and

c. Less than or equal to 14 Effective Full Power Days per calendar year.

APPLICABILITY: Modes 1* and 2*#.

ACTION:

When COLSS is in service and

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

   1. Restore the regulating CEA groups to within the limits, or

   2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using the above figure.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.4.

#With K_{eff} greater than or equal to 1.0.

ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per calendar year, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

2. Be in at least HOT STANDBY within 6 hours.

When COLSS is out of service and the regulating CEA groups are inserted beyond the Short Term Steady Insertion Limit except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

a. Restore the regulating CEA group to within the limit, or

b. Reduce thermal power to less than or equal to that fraction of Rated Thermal Power which is allowed by the CEA group position and the Short Term Steady State Insertion Limit.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The position of the part length CEA group shall be restricted to prevent the neutron absorber section of the part length CEA group from covering the same axial segment of the fuel assemblies for a period in excess of 7 EFPD out of any 30 EFPD period. The position of the part length CEA group shall also be limited to the insertion limits shown on Figure 3.1-3.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the neutron absorber section of the part length CEA group covering any same axial segment of the fuel assemblies for a period exceeding 7 EFPD out of any 30 EFPD period, either:

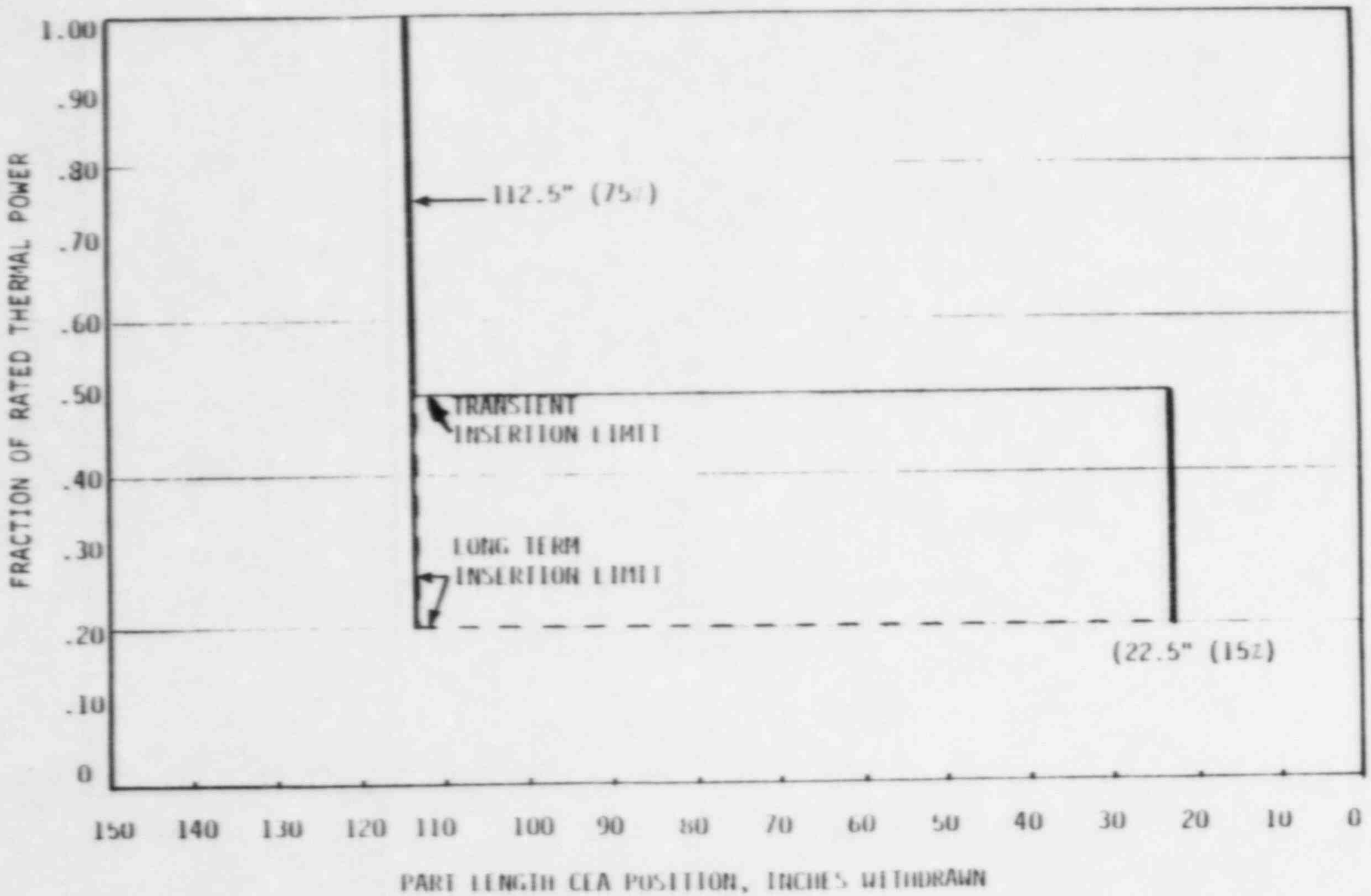
a. Reposition the part length CEA group to ensure no neutron absorber section of the part length CEA group is covering the same axial segment of the fuel assemblies within 2 hours, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA group shall be determined at least once per 12 hours.

Figure 3.1-3 PART LENGTH CEA INSERTION LIMIT vs. THERMAL POWER



ATTACHMENT C

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and regulating) CEAs, and all part length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full length or part length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one full length or part length CEA misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

*See Special Test Exceptions 3.10.2 and 3.10.4.

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ACTION: (Continued)

d. With one or more full length or part length CEAs misaligned from any other CEAs in its group by more than 7 inches but less than or equal to 19 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

e. With one full length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

f. With one full length CEA inoperable due to causes other than addressed by ACTION a. above, but within its above specified alignment requirements and either withdrawn to greater than or equal to 145 inches or within the Long Term Steady State Insertion Limits if in full length CEA group 6, operation in MODES 1 and 2 may continue.

g. With one part length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 7 inches (indicated position) of all other part length CEAs in its group.

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REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2, with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. Less than or equal to 14 Effective Full Power Days per calendar year.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

1. Restore the regulating CEA groups to within the limits, or

2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using the above figure.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

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*See Special Test Exceptions 3.10.2 and 3.10.4.

#With K_{eff} greater than or equal to 1.0.

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ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per calendar year, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

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PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The position of the part length CEA group shall be:

a. Withdrawn to $\geq$ 145" or;

b. Restricted to prevent the neutron absorber section of the part length CEA group from covering the same axial segment ($\leq$ 145") of the fuel assemblies for a period in excess of 7 EFPD out of any 30 EFPD period.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the neutron absorber section of the part length CEA group covering any same axial segment of the fuel assemblies for a period exceeding 7 EFPD out of any 30 EFPD period, either:

a. Reposition the part length CEA group to ensure no neutron absorber section of the part length CEA group is covering the same axial segment of the fuel assemblies within 2 hours, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA group shall be determined at least once per 12 hours.

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ATTACHMENT D

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and regulating) CEAs, and all part length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full length or part length CEA inoperable or misaligned from an other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one full length or part length CEA misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with Figure 3.1-1A and that within one hour the misaligned CEA is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

   2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a. Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

      b. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

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* See Special Test Exceptions 3.10.2 and 3.10.4.

ACTION: (Continued)

d. With one or more full length or part length CEAs misaligned from any other CEAs in its group by more than 7 inches but less than or equal to 19 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with Figure 3.1-1A and that within one hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

a. Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

b. The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

e. With one full length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

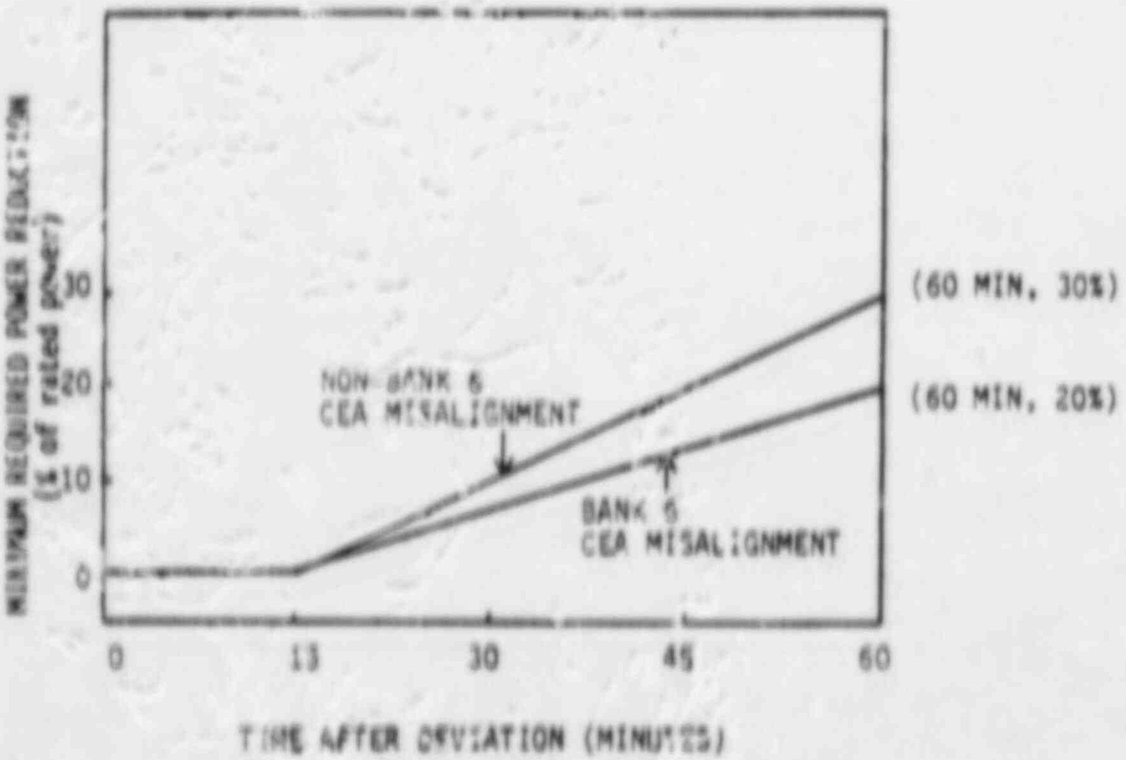
f. With one full length CEA inoperable due to causes other than addressed by ACTION a. above, but within its above specified alignment requirements and either withdrawn to greater than or equal to 145 inches or within the Long Term State Insertion Limits if in full length CEA group 6, operation in MODES 1 and 2 may continue.

g. With one part length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 7 inches (indicated position) of all other part length CEAs in its group.

Encl. (2) to
S-CE-9194

FIGURE 3.1 - 1A

Required Power Reduction after CEA Deviation*



* When core power is reduced to 60% of rated power per this limit curve, further reduction is not required by this specification.

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2, when COLSS is in service and to the Short Term Steady State Insertion Limits when COLSS is out-of-service with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. Less than or equal to 4 hours per 24 hour interval,

b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day Interval, and

c. Less than or equal to 14 Effective Full Power Days per calendar year.

APPLICABILITY: Modes 1* and 2*#.

ACTION:

When COLSS is in service and

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

   1. Restore the regulating CEA groups to within the limits, or

   2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using the above figure.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

   1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

   2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.4.

#With K_{eff} greater than or equal to 1.0.

ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per calendar year, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

2. Be in at least HOT STANDBY within 6 hours.

When COLSS is out of service and the regulating CEA groups are inserted beyond the Short Term Steady Insertion Limit except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

a. Restore the regulating CEA group to within the limit, or

b. Reduce thermal power to less than or equal to that fraction of Rated Thermal Power which is allowed by the CEA group position and the Short Term Steady State Insertion Limit.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The position of the part length CEA group shall be restricted to prevent the neutron absorber section of the part length CEA group from covering the same axial segment of the fuel assemblies for a period in excess of 7 EFPD out of any 30 EFPD period. The position of the part length CEA group shall also be limited to the insertion limits shown on Figure 3.1-3.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the neutron absorber section of the part length CEA group covering any same axial segment of the fuel assemblies for a period exceeding 7 EFPD out of any 30 EFPD period, either:

a. Reposition the part length CEA group to ensure no neutron absorber section of the part length CEA group is covering the same axial segment of the fuel assemblies within 2 hours, or

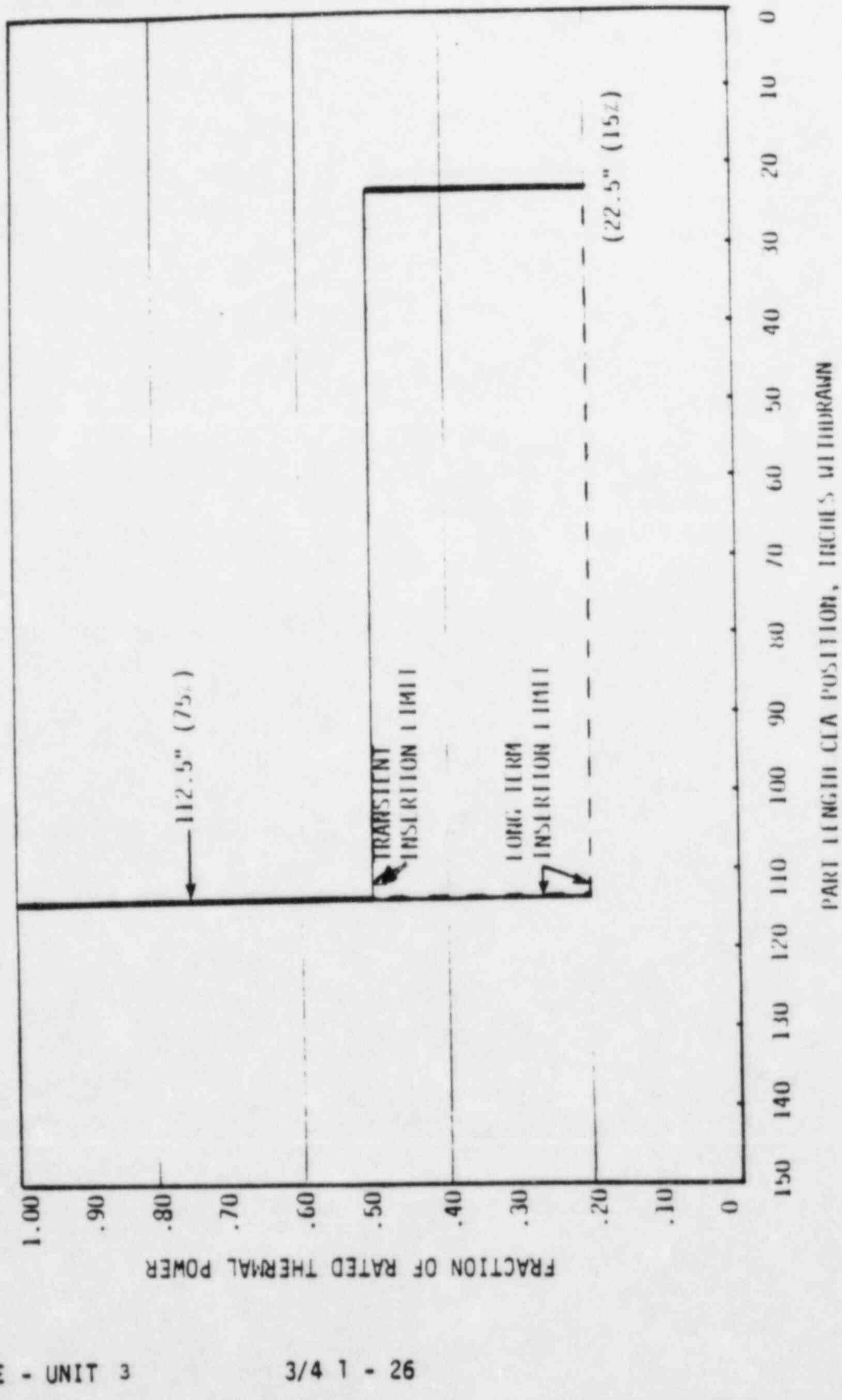
b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA group shall be determined at least once per 12 hours.

Figure 3.1-3

Figure 3.1-3 PART LENGTH CEA INSERTION LIMIT vs. THERMAL POWER



DESCRIPTION OF PROPOSED CHANGE NPF-10/15-155
AND SAFETY ANALYSIS

This is a request to add a new Technical Specification Section 3/4.4.10, REACTOR COOLANT GAS VENT SYSTEM.

Proposed Specification

Unit 2: See Attachment "A"
Unit 3: See Attachment "B"

Description

Generic Letter 83-37 dated November 1, 1983 required licensees to submit Technical Specifications (TS) for the reactor coolant system vents required by NUREG-0737, "Clarification of TMI Action Plan Requirements." Southern California Edison Company's (SCE) letter dated December 9, 1983 in response to Generic Letter 83-37, stated that SCE would submit proposed Technical Specifications for the Reactor Coolant Gas Vent System (RCGVS) by April 1, 1984. The RCGVS is described in FSAR Section 9.3.7 and illustrated in Attachment "C". The proposed Technical Specification is consistent with the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 RCGVS design and is necessarily different than the Standard Technical Specification (STS) provided with Generic Letter 83-37. The Standard Technical Specification appears to be intended for designs with single independent vent paths from the RCS vent points. It has several shortcomings when applied to designs with multiple interdependent paths such as SONGS 2 and 3. Specifically, the STS would allow one vent path to have a single valve open provided that another vent path from that vent point was operable with both valves closed. The proposed TS for SONGS 2 and 3 delineates the valves required to be operable and specifies the valves which must be closed to maintain the double isolaton intended by the STS.

The proposed ACTION statements also differ from the STS to be consistent with the SONGS 2 and 3 design. In addition the proposed actions are more flexible than the STS as follows:

1. The STS action allows 30 days of operation with a vent path inoperable. The proposed action allows until the next cold shutdown. This is justified considering the fact that the RCGVS is not required for plant cooldown or for the mitigation of any design bases accident. The RCGVS is designed such that the loss of coolant resulting from the inadvertent operation or failure of a single operable vent valve de-energized and closed in compliance with the action requirements would be within the capacity of the charging system and would only constitute RCS leakage in excess of the limits prescribed by Specification 3/4.4.5.1, RCS Leakage. Specification 3/4.4.5.1 delineates the actions t be taken when RCS leakage is greater than the allowed limits.

2. The STS does not have a 3.0.4 exception. Since most repairs of the RCGVS would require a cold shutdown, a cold shutdown would be necessitated after any trip while complying with the action before returning to MODE 1. The proposed change provides a 3.0.4 exception for entry into MODES 3, 2 and 1 to allow return to power operation following a trip, while in compliance with action requirements. As noted above, the RCGVS is not required for plant cooldown nor is it credited in the mitigation of any design bases accident. Additionally, the RCGVS is designed such that failure while complying with the action would constitute excess RCS leakage which is addressed by Specification 3/4.4.5.1. Because of this, a forced cold-shutdown following a trip while complying with the 3/4.4.11 action requirements is unnecessarily restrictive. Considering the lack of safety significance for the RCGVS and the existing provisions of the TS, the proposed 3.0.4 exception is justified.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No

   The RCGVS was designed and installed to vent non-condensible gases and/or steam from the reactor coolant system (RCS) that could inhibit natural circulation core cooling following a non-design bases accident. The RCGVS is designed to not contribute to the occurrence nor is it credited with mitigating the consequences of any previously analyzed accident. Therefore, the proposed Technical Specifications for the RCGVS will not increase the probability or consequences of any previously analyzed accident.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

Failure or inadvertent operation of the RCGVS would resemble RCS leakage in excess of the allowable limits but within the capacity of the charging system. This event has been previously evaluated and Technical Specification 3/4.4.5.1 prescribes actions to be taken in the event of RCS leakage in excess of the allowable limits. Therefore, the proposed addition of Technical Specifications for the RCGVS does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No

As stated above, the proposed change does not affect the probability or consequences of any previously evaluated accident nor does it create the possibility of a new or different kind of accident. Therefore, no margin of safety is reduced.

48 FR 14864 provided examples of amendments not likely to involve a significant hazards consideration. The proposed change described above is most similar to example (ii) in that it institutes new requirements previously not embodied in the Technical Specifications.

Safety and Significant Hazards Determination

Based on the above Safety Analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

PWS:1112F

ATTACHMENT "A"

REACTOR COOLANT SYSTEM

REACTOR COOLANT GAS VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.10 The Reactor Coolant Gas Vent System shall be OPERABLE with:

a. At least one of valves 2HV0296A or 2HV0296B capable of being powered from an emergency bus and providing a vent path from the reactor vessel head; and,

b. At least one of valves 2HV0297A or 2HV0297B capable of being powered from an emergency bus and providing a vent path from the pressurizer steam space; and,

c. At least one of valves 2HV0299-1, capable of being powered from an emergency bus and providing a vent path to the containment atmosphere, or 2HV0298-2, capable of being powered from an emergency bus and providing a vent path to the quench tank; and

d. Valves 2HV0296A, 2HV0296B, 2HV0297A or 2HV0297B, 2HV0299-1 and 2HV0298-2 all closed.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any of valves 2HV0296A, 2HV0296A, 2HV0926B, 2HV0297A or 2HV0297B inoperable, operation may continue provided that:

   i) power is removed from the inoperable valve(s); and,

   ii) valves 2HV0299-1 and 2HV0298-2 are maintained closed with power removed; and,

   iii) the inoperable valve(s) is restored to OPERABLE status during the next COLD SHUTDOWN.

b. With either of valves 2HV0299-1 or 2HV0298-2 inoperable, operation may continue provided that:

   i) power is removed from the inoperable valve; and

   ii) valves 2HV0296A, 2HV0296B, 2HV0297A and 2HV0297B are all maintained closed with power removed; and

LIMITING CONDITION FOR OPERATION

iii) the inoperable valve(s) is restored to OPERABLE status during during the next COLD SHUTDOWN.

c. The provisions of 3.0.4 are not applicable for entry into MODES 3, 2 and 1.

SURVEILLANCE REQUIREMENTS

4.4.10 Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

3. Verifying flow through the reactor coolant vent system vent paths during venting during COLD SHUTDOWN or REFUELING.

Add the following to the BASES section:

REACTOR COOLANT SYSTEM

BASES

3/4.4.10 REACTOR COOLANT GAS VENT SYSTEM

Reactor coolant system gas vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling following a non-design bases accident. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The design redundancy of the Reactor Coolant Gas Vent System serves to minimize the probaiblity of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant Gas Vent System are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

PWS:1112F

ATTACHMENT "B"

REACTOR COOLANT GAS VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.10 The Reactor Coolant Gas Vent System shall be OPERABLE with:

a. At least one of valves 3HV0296A or 3HV0296B capable of being powered from an emergency bus and providing a vent path from the reactor vessel head; and,

b. At least one of valves 3HV0297A or 3HV0297B capable of being powered from an emergency bus and providing a vent path from the pressurizer steam space; and,

c. At least one of valves 3HV0299-1, capable of being powered from an emergency bus and providing a vent path to the containment atmosphere, or 3HV0298-2, capable of being powered from an emergency bus and providing a vent path to the quench tank; and

d. Valves 3HV0296A, 3HV0296B, 3HV0297A or 3HV0297B, 3HV0299-1 and 3HV0298-2 all closed.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any of valves 3HV0296A, 3HV0296A, 3HV0926B, 3HV0297A or 3HV0297B inoperable, operation may continue provided that:

   i) power is removed from the inoperable valve(s); and,

   ii) valves 3HV0299-1 and 3HV0298-2 are maintained closed with power removed; and,

   iii) the inoperable valve(s) is restored to OPERABLE status during the next COLD SHUTDOWN.

b. With either of valves 3HV0299-1 or 3HV0298-2 inoperable, operation may continue provided that:

   i) power is removed from the inoperable valve; and

   ii) valves 3HV0296A, 3HV0296B, 3HV0297A and 3HV0297B are all maintained closed with power removed; and

REACTOR COOLANT SYSTEM

REACTOR COOLANT GAS VENT SYSTEM

LIMITING CONDITION FOR OPERATION

iii) the inoperable valve(s) is restored to OPERABLE status during during the next COLD SHUTDOWN.

c. The provisions of 3.0.4 are not applicable for entry into MODES 3, 2 and 1.

SURVEILLANCE REQUIREMENTS

4.4.10 Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

3. Verifying flow through the reactor coolant vent system vent paths during venting during COLD SHUTDOWN or REFUELING.

Add the following to the BASES section:

REACTOR COOLANT SYSTEM

BASES

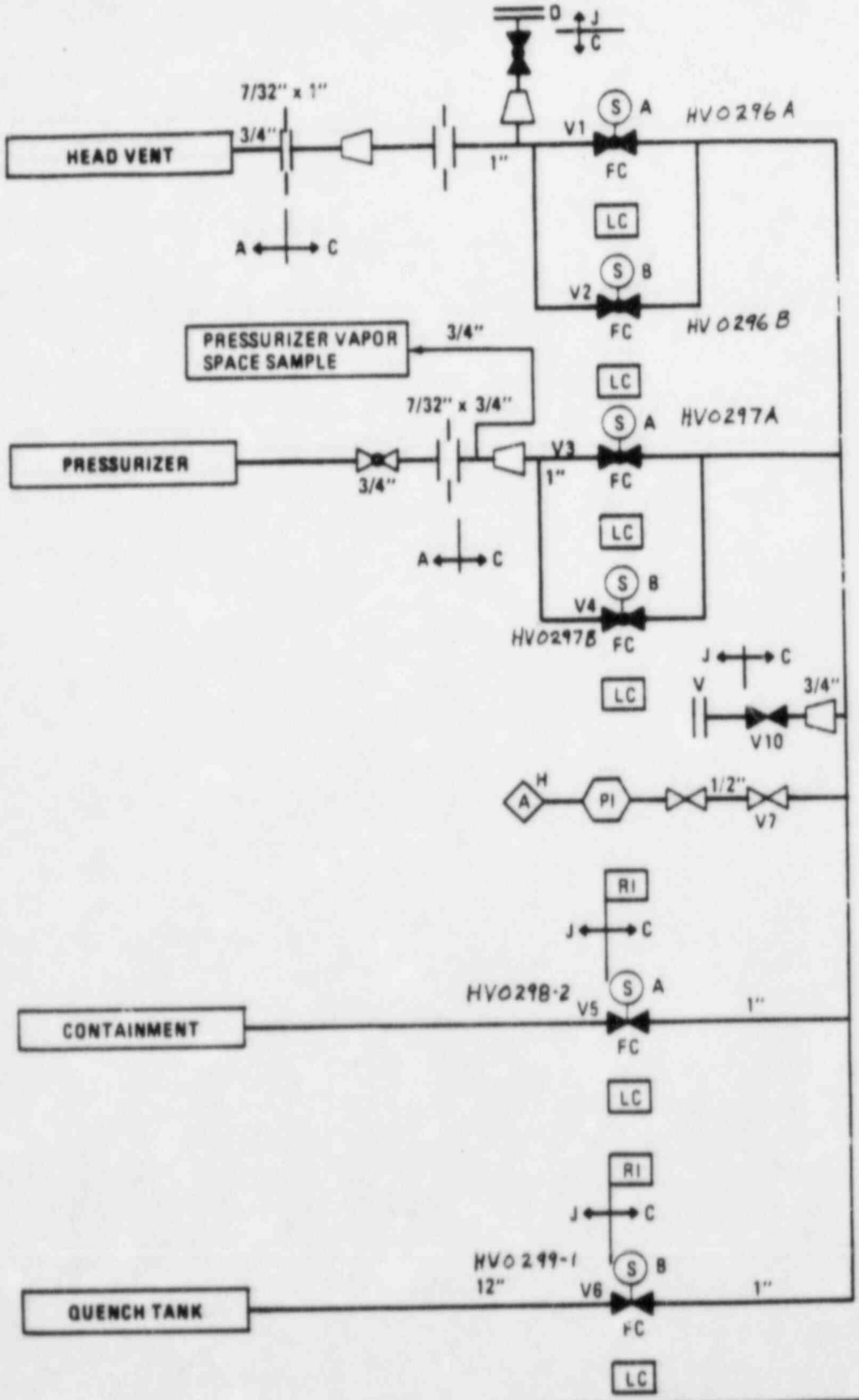
3/4.4.10 REACTOR COOLANT GAS VENT SYSTEM

Reactor coolant system gas vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling following a non-design bases accident. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The design redundancy of the Reactor Coolant Gas Vent System serves to minimize the probaiblity of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant Gas Vent System are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

ATTACHMENT "C"



Updated

SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

REACTOR COOLANT GAS VENT
SYSTEM SKETCH

Figure 9.3-15