SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

	Valve Number	Valve Function	Valve Position
a.	HV9353	SDC Warmup	CLOSED
b.	HV9359	SDC Warmup	CLOSED
c.	HV8150	SDC(HX) Isolation	CLOSED
d.	HV8151	SDC(HX) Isolation	CLOSED
e.	HV8152	SDC(HX) Isolation	CLOSED
f.	HV8153	SDC(HX) Isolation	CLOSED
g.	HV0396	SDC Bypass Flow Control	CLOSED
h.	HV8161	SDC(HX) Bypass Flow Isolation	OPEN
i.	HV9420	Hot Leg Injection Isolation	CLOSED
j.	HV9434	Hot Leg Injection Isolation	CLOSED
k.	HV8160	SDC Bypass Flow Control	OPEN
l.	10-068	RWST Isolation	LOCKED OPEN (MANUAL)
m.	HV8162	LPSI Miniflow Isolation	OPEN
n.	HV8163	LPSI Miniflow Isolation	OPEN

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2. 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 the correct position.

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

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

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when RCS pressure is simulated greater than or equal to 715 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when simulated RCS pressure is greater than or equal to 376 psia.

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ATTACHMENT B

(Existing Technical Specification)

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

	Valve Number	Valve Function	Valve Position
a.	HV9353	SDC Warmup	CLOSED
b.	HV9359	SDC Warmup	CLOSED
c.	HV8150	SDC(HX) Isolation	CLOSED
d.	HV8151	SDC(HX) Isolation	CLOSED
e.	HV8152	SDC(HX) Isolation	CLOSED
f.	HV8153	SDC(HX) Isolation	CLOSED
g.	HV0396	SDC Bypass Flow Control	CLOSED
h.	HV8161	SDC(HX) Bypass Flow Isolation	OPEN
i.	14-081	HV-0396 Isolation	LOCKED OPEN (MANUAL)
j.	14-082	HV-0396 Isolation	LOCKED OPEN (MANUAL)
k.	HV9420	Hot Leg Injection Isolation	CLOSED
l.	HV9434	Hot Leg Injection Isolation	CLOSED
m.	HV8160	SDC Bypass Flow Control	OPEN
n.	10-068	RWST Isolation	LOCKED OPEN (MANUAL)
o.	HV8162	LPSI Miniflow Isolation	OPEN
p.	HV8163	LPSI Miniflow Isolation	OPEN

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2. 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 the correct position.

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

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

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when RCS pressure is simulated greater than or equal to 715 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when simulated RCS pressure is greater than or equal to 376 psia.

ATTACHMENT C

(Proposed Technical Specification)

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
a. HV9353	SDC Warmup	CLOSED
b. HV9359	SDC Warmup	CLOSED
c. HV8150	SDC(HX) Isolation	CLOSED
d. HV8151	SDC(HX) Isolation	CLOSED
e. HV8152	SDC(HX) Isolation	CLOSED
f. HV8153	SDC(HX) Isolation	CLOSED
g. HV0396	SDC Bypass Flow Control	CLOSED
h. HV8161	SDC(HX) Bypass Flow Isolation	OPEN
i. HV9420	Hot Leg Injection Isolation	CLOSED
j. HV9434	Hot Leg Injection Isolation	CLOSED
k. HV8160	SDC Bypass Flow Control	OPEN
l. HV8162	LPSI Miniflow Isolation	OPEN
m. HV8163	LPSI Miniflow Isolation	OPEN

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and
2. 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 the correct position.

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

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and
2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when RCS pressure is simulated greater than or equal to 715 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when simulated RCS pressure is greater than or equal to 376 psia.

ATTACHMENT D

(Proposed Technical Specification)

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

	Valve Number	Valve Function	Valve Position
a.	HV9353	SDC Warmup	CLOSED
b.	HV9359	SDC Warmup	CLOSED
c.	HV8150	SDC(HX) Isolation	CLOSED
d.	HV8151	SDC(HX) Isolation	CLOSED
e.	HV8152	SDC(HX) Isolation	CLOSED
f.	HV8153	SDC(HX) Isolation	CLOSED
g.	HV0396	SDC Bypass Flow Control	CLOSED
h.	HV8161	SDC(HX) Bypass Flow Isolation	OPEN
i.	14-081	HV-0396 Isolation	LOCKED OPEN (MANUAL)
j.	14-082	HV-0396 Isolation	LOCKED OPEN (MANUAL)
k.	HV9420	Hot Leg Injection Isolation	CLOSED
l.	HV9434	Hot Leg Injection Isolation	CLOSED
m.	HV8160	SDC Bypass Flow Control	OPEN
n.	HV8162	LPSI Miniflow Isolation	OPEN
o.	HV8163	LPSI Miniflow Isolation	OPEN

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and
2. 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 the correct position.

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

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and
2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when RCS pressure is simulated greater than or equal to 715 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when simulated RCS pressure is greater than or equal to 376 psia.

DESCRIPTION OF PROPOSED CHANGE NPF-10/15-207 AND SAFETY ANALYSIS

This is a request to revise Technical Specifications 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation," 3/4.6.3, "Containment Isolation Valves" and 3/4.7.1.5, "Main Steam 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

The proposed change revises Technical Specifications (TS) 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation (ESFAS)," 3/4.7.1.5, "Main Steam Isolation Valves (MSIV's)," and 3/4.6.3, "Containment Isolation Valves." TS 3/4.3.2 specifies the number of channels and type of ESFAS instrumentation required to be operable, response times and periodic surveillance tests to verify operability, and actions to be taken when the minimum operability requirements are not met. TS 3/4.7.1.5 defines operability requirements for MSIV's and actions to be taken when one or both MSIV's are inoperable. TS 3/4.6.3 specifies operability requirements for containment isolation valves surveillance requirements and actions to be taken when operability requirements are not met. The operability requirements for the Main Steam Isolation Valves ensures that no more than one steam generator will blow down in the event of a main steam line rupture assuming a single failure. Ensuring that only one steam generator blows down prevents the containment design pressure from being exceeded and limit positive reactivity addition due to cooldown of the reactor coolant system.

During normal plant operation, the MSIV's are maintained open by hydraulic pressure working against compressed nitrogen gas. The energy stored in the compressed gas provides the motive force for valve closure. Technical Specifications currently require an MSIV closure time of 5.0 seconds. The pressure required to maintain the valve open and provide a 5.0 second response time is high. Dynamic effects on components in the MSIV hydraulic circuits, due in part to the high pressures, have resulted in component failures and spurious MSIV closures during plant operation. A spurious MSIV closure during power operation will result in a reactor trip. Reduction in the MSIV operating pressure will result in increased component reliability but necessitate a slower MSIV response time.

The proposed change increases MSIV closure time from 5.0 to 8.0 seconds. Specifically, the response time listed for the MSIV's in Table 3.3-5, "ESFAS Response Times" under mainsteam Isolation Signal (MSIS) is increased from 5.9 to 8.9 seconds (0.9 seconds is allowed for instrumentation response time, the remainder for the valve). The response times for the MSIV's listed in Table 3.6-1, "Containment Isolation Valves is increased from 5.0 to 8.0 seconds. Likewise, the response time included in TS 3/4.7.15 is increased from 5.0 to 8.0 seconds.

Safety Analysis

The proposed changes discussed 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?

   Response: No

   The proposed change affects the closure time of the main steam isolation valves. Main Steam Isolation Valves are provided to prevent the blowdown of more than one steam generator in the event of a main steam line break.

   Thus, the proposed change potentially could affect the probability or consequences of the Main Steam Line Break (MSLB) accident which has been previously evaluated. The consequences of a MSLB manifest themselves in rapid RCS cooldown, potentially resulting in a post trip return to power in the presence of a negative moderator temperature coefficient, with the potential for exceeding specified acceptable fuel design limits and, for an MSLB inside containment, the potential for containment overpressurization. Both aspects of the MSLB have been reanalyzed.

   The effect of a longer MSIV closure time on RCS cooldown and possible return to power following a MSLB event is included in the Cycle 3 reload analyses, which assumed a 10 second MSIV closure time. The results of the analysis of the limiting MSLB (from Hot Full Power) predicts no fuel pin failures; therefore, a coolable geometry is maintained and the acceptance criteria are satisfied. In addition, the limiting (from Hot Full Power) Cycle 2 case was reanalyzed assuming a 10 second closure time. The limiting analysis for Cycle 2 predicted no fuel pin failures. The increase in MSIV response time did not have a significant impact on this event. These analyses are conservative since the proposed change increases the response to 8 seconds whereas 10 seconds is assumed in the analyses.

   For the MSLB inside containment, the worst case (i.e. resulting in the peak containment pressure) is from hot full power with loss of offsite power and one train of containment cooling. With the current response time, the peak containment pressure is 55.7 psig. The peak containment pressure is not expected to be greater than 55.7 psig with an 8.0 second response time because mass/energy releases to containment with the new response time are bounded by the original FSAR analysis. This results from more realistic and detailed modeling than that used in the original FSAR analysis. The significant changes to the FSAR methodology were:

(i) The steam lines were modeled as two separate nodes, associated with each steam generator instead of being combined with the steam generator node.

(ii) Three separate flow resistances (from each steam generator, to the cross tie, and for the cross tie itself) were used instead of a single, combined flow resistance.

(iii) The actual MSIV closure characteristics were used instead of a step change closure.

(iv) The feed flow split between the ruptured and intact units was assumed to be 175% and 25% (of full flow) to the ruptured and intact units respectively, rather than the overly conservative 200% and 0% assumed in the previous analyses.

(v) The Darcy equation, used to calculate steam line flow, was modified to account for compressibility.

(vi) Choking at various points in the steam lines was considered when appropriate conditions existed.

Because of these model enhancements, the peak containment pressure with MSIV response time increase to 8.0 seconds remains bounded by the current 55.7 psig. Therefore, the proposed change does not significantly increase the probability or consequences of the MSLB event.

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 does not affect the configuration of the facility or the manner in which it is operated. Therefore, the proposed change 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 proposed change does not reduce the effectiveness of the main steam isolation valves. Analysis has demonstrated that the increased closure time has a negligible impact on safety analysis results. Therefore, no margin of safety is reduced.

The Commission has provided guidance for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase in the probability or consequences of a previously-analyzed accident or may in some way reduce a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP). In this case, SRP Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," and SRP Section 15.1.5, "Steam System Piping Failures Inside and Outside Containment" delineate the pertinent acceptance criteria for the analyses of MSLB events. SRP Section 6.2.4 requires that mass and energy releases from postulated secondary system pipe ruptures be considered to assure that the containment design margin is maintained. SRP Section 15.1.5 requires that the capability to cool the core be maintained throughout the event.

The proposed increase in MSIV response time to 8.0 seconds has been analyzed. The results show that the capability to cool the core is maintained with the proposed increase in MSIV response time. Therefore, the proposed change satisfies SRP 15.1.4 acceptance criteria.

In addition, the containment response to the limiting MSLB inside containment was analyzed with the proposed increase MSIV response time. The results showed that the peak containment pressure is bounded by the current analysis (which calculates a peak containment pressure of 55.7 psig). The new analysis incorporates enhancements to more realistically model the event. The improvements include the use of a two node model for the steam lines, calculation of separate flow resistances between nodes, crediting choking of steam flow when conditions merit, use of actual MSIV flow characteristics, accounting for compressible flow and a revised feed flow split between the affected and unaffected steam generator. With these enhancements, the proposed change results in mass and energy releases to the containment which are bounded by the previous analysis and maintains the containment design margin. Therefore, the proposed change satisfies the SRP acceptance criteria and is similar to Example (vi) of 48 FR 14870.

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

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.

ATTACHMENT A

Unit 2 Existing Specifications

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	5.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV4054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kv Emergency Bus Undervoltage	
LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
9. Steam Generator Level - Low (and ΔP - High)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
10. Control Room Ventilation Airborne Radiation	
CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)

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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

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subseqent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5.0 seconds when tested pursuant to Specification 4.0.5.

ATTACHMENT B

Unit 2 Proposed Specifications

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	8.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV4054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kv Emergency Bus Undervoltage	
LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
9. Steam Generator Level - Low (and ΔP - High)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)
10. Control Room Ventilation Airborne Radiation	
CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)

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	8
33	HV-8205#	Mainsteam isolation	8
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

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subseqent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 8.0 seconds when tested pursuant to Specification 4.0.5.

ATTACHMENT C

Unit 3 Existing Specifications

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
a. MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	5.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV5054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
a. RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kV Emergency Bus Undervoltage	
a. LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
9. Steam Generator Level - Low (and ΔP - High)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
10. Control Room Ventilation Airborne Radiation	
a. CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	36 (NOTE 5)

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

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subseqent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5.0 seconds when tested pursuant to Specification 4.0.5.

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ATTACHMENT D

Unit 3 Proposed Specifications

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
a. MSIS	
(1) Main Steam Isolation (HV8204, HV8205)	8.9
(2) Main Feedwater Isolation (HV4048, HV4052)	10.9
(3) Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV5054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9
(4) Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9
6. Refueling Water Storage Tank - Low	
a. RAS	
(1) Containment Sump Valves Open	50.7*
7. 4.16 kV Emergency Bus Undervoltage	
a. LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
9. Steam Generator Level - Low (and ΔP - High)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2) Auxiliary Feedwater (Steam/DC train)	42.7 (Note 6)
10. Control Room Ventilation Airborne Radiation	
a. CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	36 (NOTE 5)

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	8
33	HV-8205#	Mainsteam isolation	8
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

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, reduce power to less than or equal to 5 percent RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subseqent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 8.0 seconds when tested pursuant to Specification 4.0.5.

DESCRIPTION OF PROPOSED CHANGES NPF-10/15-209 AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.9.6, "Refueling Machine".

Existing Technical Specifications

Unit 2: See Attachment A

Unit 3: See Attachment C

Proposed Technical Specifications

Unit 2: See Attachment B

Unit 3: See Attachment D

Description

The proposed change revises Technical Specification 3/4.9.6, "Refueling Machine". Technical Specification 3/4.9.6 requires that the refueling machine be used for movement of Control Element Assemblies (CEAs) or fuel assemblies and be operable within specified weight limits. The action to be taken is also prescribed therein when the refueling machine becomes inoperable. The surveillance requirements further require that a load test within specified weight limits be performed to ensure the operability of the refueling machine prior to the start of any intended operations.

The proposed change to Technical Specification 3/4.9.6 revises the existing Limiting Condition for Operation (LCO) with inclusion of the refueling machine auxiliary hoist and a corresponding ACTION statement to suspend its operations when the specified LCO requirement is not met. The change will allow the use of the refueling machine auxiliary hoist for movement of CEAs and a specially designed four (4) or five (5) finger CEA lift tool so long as the hoist is operable with an overload cut off limit of less than or equal to 1000 pounds. Applicability of the refueling machine auxiliary hoist and refueling machine for movement of CEAs and/or fuel assemblies has been redefined. The ACTION statement further imposes an operating restraint to any operations which do not comply with this overload cut off limit in parallel to that applicable to the refueling machine in the present version of Technical Specification 3/4.9.6. The change also revises Surveillance Requirements 4.9.6 to reflect such compliance with the operation of the refueling machine auxiliary hoist in addition to the existing one for the refueling machine. The technical specification bases are further modified to include the functional division of the refueling machine and refueling machine auxiliary hoist embodied in LCO.

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 an accident previously evaluated?

   Response: No

   The probability or consequence of an accident is not increased by the proposed change since the refueling machine auxiliary hoist meets all the design criteria for CEA handling equipment specified in the Final Safety Analysis Report (FSAR) for SONGS Units 2 and 3 and the requirements of NUREG-0612. Thus, this proposed change will not involve a significant increase in the probability of an accident previously evaluated.

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

   Response: No

   The refueling machine auxiliary hoist is equipped with interlocks which prevent its concurrent use of the refueling machine. Additionally, it also has mechanical interlocks to secure the CEA during a shuffle and will therefore not distort or damage such a CEA while it is being moved under water. Since the refueling machine auxiliary hoist will not be used to move fuel assemblies, the operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   There is no reduction in the margin of safety previously established, since the operation of the refueling machine auxiliary hoist under the proposed LCO condition will not present any increased potential for damage to CEAs, nor will it affect the existing safety analyses and design criteria.

Insert X

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered least likely to involve significant hazards considerations. Example (ii) from the Federal Register relates to a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications would not be likely to involve significant hazards considerations. The proposed change is similar to Example (ii) in that it incorporates an additional control for the operation of the refueling machine auxiliary hoist not presently included in Technical Specification 3/4.9.6.

The existing technical specification and its bases were written without consideration to account for modifications incorporating the operation of the refueling machine auxiliary hoist. When installed on the refueling machine, the refueling machine auxiliary hoist can be used to facilitate the CEA change over a reactor core with a specially designed four (4) or five (5) finger lift tool in addition to the existing CEA change mechanism described in FSAR Section 9.1.4.2.2.5. Furthermore, it can be used to handle refueling equipment, to perform coupling and uncoupling of CEA extension shafts, and to verify such intended coupling and uncoupling. The proposed change will then restrict the operation of the refueling machine to movement of fuel assemblies with or without CEAs and that of the refueling machine auxiliary hoist to movement of CEAs, respectively. It will also incorporate the applicable LCO requirement for the operation of the refueling machine auxiliary hoist to handle these functions pertinent to CEA movements only. Additionally, the revision to the corresponding ACTION, Surveillance Requirements, and Bases sections extends the compliance with such an LCO requirement for the refueling machine auxiliary hoist in conjunction with those applied to the refueling machine. In short, the proposed change manifests the functional separation of fuel assembly and CEA movements respectively in APPLICABILITY by use of the refueling machine and refueling machine auxiliary hoist with a corresponding but similar set of LCO, ACTION, and Surveillance Requirements statements. Since the refueling machine auxiliary hoist can be used in conjunction with a specially designed four (4) finger lift tool to accommodate a four (4) finger CEA movement, the restriction identified in the footnote of Technical Specification 3/4.9.6 has been deleted accordingly.

The refueling machine auxiliary hoist meets all the design criteria for CEA handling equipment specified in FSAR Section 9.1.4.1.3.1 and is equipped with interlocks to prevent its concurrent use with the refueling machine. Load cells are included to limit the maximum permissible load to 1000 pounds. Further, the four (4) or five (5) finger CEA handling tool to be used in conjunction with the refueling machine auxiliary hoist has mechanical interlocks to secure the CEA during a shuffle and is long enough to preclude raising a CEA above minimum safe water cover depth. Consequently, the refueling machine auxiliary hoist would not be likely to distort or damage a CEA while it is being moved. This is because the CEA is always shuffled under water where acceleration and swinging are dampened even at full hoist speed. In addition, the existing safety analyses pertaining to fuel handling accidents would not be affected since the refueling machine auxiliary hoist is not intended for moving fuel assemblies with or without CEAs. More important, the operation of the refueling machine auxiliary hoist under the specified LCO condition will meet NUREG-0612 requirements for control of heavy loads at nuclear power plants. Because the operation of the subject refueling machine auxiliary hoist would not affect the existing safety analyses and meet design criteria and NUREG-0612 requirements, the proposed change to incorporate control of such an operation is similar to Example (ii) of 48 FR 14870.

Safety and Significant Hazards Determination

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. 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 Environment Statement.

ATTACHMENT A

Existing Technical Specifications, Unit 2

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs* or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3000 pounds, and

b. An overload cut off limit of less than or equal to 3350 pounds.

APPLICABILITY: During movement of CEAs* and/or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for the refueling machine OPERABILITY not satisfied, suspend all refueling machine operations involving the movement of CEAs* and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine used for movement of CEAs* or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3000 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 3350 pounds.

*Except for movement of four finger CEA's, coupling and uncoupling the CEA extension shafts or verifying the coupling and uncoupling.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of all fuel assemblies including those with a CEA inserted, (2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

With the exception of the four finger CEA's, CEA's are removed from the reactor vessel along with the fuel bundle in which they are inserted utilizing the refueling machine. The four finger CEA's are inserted through the upper guide structure with two fingers in each of the two adjacent fuel bundles in the periphery of the core. The four finger CEA's are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane.

Coupling and uncoupling of the CEA's and the CEDM drive shaft extensions is accomplished using the gripper operating tool. The coupling and uncoupling is verified by weighing the drive shaft extensions.

ATTACHMENT B

Proposed Technical Specifications, Unit 2

REFUELING OPERATIONS

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of fuel assemblies with or without CEAs and shall be OPERABLE with:

a. A minimum capacity of 3000 pounds, and

b. An overload cut off limit of less than or equal to 3350 pounds.

The refueling machine auxiliary hoist may be used for the movement of CEAs without fuel bundles and shall be OPERABLE with an overload cut off limit of less than or equal to 1000 pounds.

APPLICABILITY: During movement of CEAs and/or fuel assemblies within the reactor pressure vessel utilizing the refueling machine auxiliary hoist or refueling machine.

ACTION: With the requirements for the refueling machine OPERABILITY not satisfied, suspend all refueling machine operations involving the movement of fuel assemblies with or without CEAs within the reactor pressure vessel. With the requirements for the refueling machine auxiliary hoist not satisfied, suspend all refueling machine auxiliary hoist operations involving the movement of CEAs within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine used for movement of fuel assemblies with or without CEAs within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3000 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 3350 pounds. The refueling machine auxiliary hoist used for movement of CEAs within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by demonstrating an automatic load cut off when the auxiliary hoist load exceeds 1000 pounds.

BASES

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of all fuel assemblies including those with a CEA inserted, (2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

Five finger CEAs are removed from the reactor vessel either along with the associated fuel bundle utilizing the refueling machine or can be removed without the associated fuel bundle utilizing the refueling machine auxiliary hoist. The four finger CEAs are inserted through the upper guide structure with two fingers in each of the two adjacent fuel bundles in the periphery of the core. The four finger CEAs are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane or the refueling machine auxiliary hoist.

Coupling and uncoupling of the CEAs and the CEDM drive shaft extensions is accomplished using one of the gripper operating tools. The coupling and uncoupling is verified by weighing the drive shaft extensions.

ATTACHMENT C

Existing Technical Specifications, Unit 3

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs* or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 3000 pounds, and

b. An overload cut off limit of less than or equal to 3350 pounds.

APPLICABILITY: During movement of CEAs* and/or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for the refueling machine OPERABILITY not satisfied, suspend all refueling machine operations involving the movement of CEAs* and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine used for movement of CEAs* or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3000 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 3350 pounds.

*Except for movement of four finger CEA's, coupling and uncoupling the CEA extension shafts or verifying the coupling and uncoupling.

BASES

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of all fuel assemblies including those with a CEA inserted, (2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

With the exception of the four finger CEA's, CEA's are removed from the reactor vessel along with the fuel bundle in which they are inserted utilizing the refueling machine. The four finger CEA's are inserted through the upper guide structure with two fingers in each of the two adjacent fuel bundles in the periphery of the core. The four finger CEA's are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane.

Coupling and uncoupling of the CEA's and the CEDM drive shaft extensions is accomplished using the gripper operating tool. The coupling and uncoupling is verified by weighing the drive shaft extensions.

ATTACHMENT D

Proposed Technical Specifications, Unit 3

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of fuel assemblies with or without CEAs and shall be OPERABLE with:

a. A minimum capacity of 3000 pounds, and

b. An overload cut off limit of less than or equal to 3350 pounds.

The refueling machine auxiliary hoist may be used for the movement of CEAs without fuel bundles and shall be OPERABLE with an overload cut off limit of less than or equal to 1000 pounds.

APPLICABILITY: During movement of CEAs and/or fuel assemblies within the reactor pressure vessel utilizing the refueling machine auxiliary hoist or refueling machine.

ACTION: With the requirements for the refueling machine OPERABILITY not satisfied, suspend all refueling machine operations involving the movement of fuel assemblies with or without CEAs within the reactor pressure vessel. With the requirements for the refueling machine auxiliary hoist not satisfied, suspend all refueling machine auxiliary hoist operations involving the movement of CEAs within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The refueling machine used for movement of fuel assemblies with or without CEAs within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 3000 pounds and demonstrating an automatic load cut off when the refueling machine load exceeds 3350 pounds. The refueling machine auxiliary hoist used for movement of CEAs within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by demonstrating an automatic load cut off when the auxiliary hoist load exceeds 1000 pounds.

BASES

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: (1) the refueling machine will be used for movement of all fuel assemblies including those with a CEA inserted, (2) each machine has sufficient load capacity to lift a fuel assembly including those with a CEA, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

Five finger CEAs are removed from the reactor vessel either along with the associated fuel bundle utilizing the refueling machine or can be removed without the associated fuel bundle utilizing the refueling machine auxiliary hoist. The four finger CEAs are inserted through the upper guide structure with two fingers in each of the two adjacent fuel bundles in the periphery of the core. The four finger CEAs are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane or the refueling machine auxiliary hoist.

Coupling and uncoupling of the CEAs and the CEDM drive shaft extensions is accomplished using one of the gripper operating tools. The coupling and uncoupling is verified by weighing the drive shaft extensions.

3/4.6.2.2 RECIRCULATION FLOW PH CONTROL SYSTEM

The operability of the recirculation flow pH control system ensures that there is sufficient trisodium phosphate available in containment to guarantee a sump pH of ≥ 7.0 during the recirculation phase of a postulated LOCA. This pH level is required to minimize the potential for chloride stress corrosion of austentitic stainless steel. The specified amount of TSP will result in a recirculation phase pH of approximately 7.2 assuming complete dissolution and maximum allowed boric acid concentrations from the borated water sources. Similarly, surveillance 4.6.2.2 will produce a pH of approximately 7.2. The specified temperature of 120 ± 10 degrees-F for the surveillance is consistent with expected long term recirculation phase sump temperature reported in the FSAR.

DESCRIPTION OF PROPOSED CHANGE
NPF-10-212 AND NPF-15-212
AND SAFETY ANALYSIS

This is a request to revise Specification 3/4.7.6, "Snubbers" of the Technical Specifications for San Onofre Nuclear Generating Station Units 2 and 3.

Existing Technical Specifications

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

Proposed Technical Specifications

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

Description

Modifications to the existing San Onofre Units 2 and 3 Technical Specifications are hereby proposed to provide clarification that visual inspections shall verify that fasteners at both snubber ends are secure and to revise the schedule for inspection of all snubbers attached to sections of safety systems which have experienced unexpected, potentially damaging transients from only during refueling outages to within six months following determination that such an event has occurred.

1) Specification 4.7.6.c - Visual Inspection Acceptance Criteria

The first sentence of the existing specification is the following:

> Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure.

The proposed change provides clarification of item (2) that the visual inspections shall verify that fasteners at both snubber ends are secure by adding a third surveillance acceptance criteria so that the first sentence of Visual Inspection Acceptance Criteria will be the following:

> Visual inspections shall verify that (1) there are no visible indications of damage or impaired OPERABILITY, and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to (a) the component or pipe and (b) the snubber anchorage are secure.

2) Specification 4.7.6.j - Refueling Outage Inspections

The first sentence of the existing specification is the following:

> During each refueling outage an inspection shall be performed of snubbers attached to sections of safety systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems.

The proposed change revises the schedule for transient event inspections from only during refueling outages to within 6 months following a determination that an unexpected potentially damaging transient has occurred. Specification 4.7.6.j will be titled and the first sentence will be the following:

> 4.7.6.j <u>Transient Event Inspections</u>
>
> An inspection shall be performed on all hydraulic and mechanical snubbers attached to sections of safety systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following a determination that such an event has occurred.

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 proposed change maintains the same operability requirements as the existing Technical Specification, thus there is no 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

   The proposed change maintains the same operability requirements as the existing Technical Specification, thus it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   The proposed change maintains the same operability requirements as the existing Technical Specifications, thus there is no reduction in a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (ii) relates to a change that constitutes an additional limitation, restriction or control not presently included in the technical specifications: for example a more stringent surveillance requirement. The proposed change is representative of example (ii) in that it adds additional controls in surveillance requirements which are in excess of the requirements of NRC Generic Letter 84-13.

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

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. 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 Environmental Statement.

TDM:5345F

ATTACHMENT A

EXISTING TECHNICAL SPECIFICATION 3/4.7.6

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 2

SURVEILLANCE REQUIREMENTS (Continued)

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.6.e or 4.7.6.f, as applicable. However, when a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

*The inspection interval shall not be lengthened more than one step at a time.

#The provisions of Specification 4.0.2 are not applicable.

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SURVEILLANCE REQUIREMENTS (Continued)

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.6.e. or 4.7.6.f. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptace criteria subsequent to their most recent service, and the functional test must have been performed within 12 month before being installed in the unit.

i. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

j. Refueling Outage Inspections

During each refueling outage an inspection shall be performed of snubbers attached to sections of safety systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems. In addition to satisfying the visual inspection acceptance criteria, freedom of motion of mechanical snubbers shall be verified using one of the following: (i) manually induced snubber movement; (ii) evaluation of in-place snubber piston setting; (iii) stroking the mechanical snubber through its full range of travel.

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ATTACHMENT B

EXISTING TECHNICAL SPECIFICATION 3/4.7.6

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 3

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.6.e or 4.7.6.f, as applicable. However, when a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

d. Functional Tests*

At least once per 18 months during shutdown, a representative sample of at least 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber of a type of that does not meet the functional test acceptance criteria of Specification 4.7.6.e or 4.7.6.f, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle
2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine motor, etc.)
3. Snubbers within 10 feet of the discharge from safety relief valve.

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

i. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

j. Refueling Outage Inspections

During each refueling outage an inspection shall be performed of snubbers attached to sections of safety systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems. In addition to satisfying the visual inspection acceptance criteria, freedom of motion of mechanical snubbers shall be verified using one of the following: (i) manually induced snubber movement; (ii) evaluation of in-place snubber piston setting; (iii) stroking the mechanical snubber through its full range of travel.

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ATTACHMENT C

PROPOSED TECHNICAL SPECIFICATION 3/4.7.6

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 2

SURVEILLANCE REQUIREMENTS (Continued)

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to (a) the component or pipe and (b) the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.6.e or 4.7.6.f, as applicable. However, when a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

*The inspection interval shall not be lengthened more than one step at a time.

#The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.6.e or 4.7.6.f for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

j. Transient Event Inspections

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of safety systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following a determination that such an event has occurred. In addition to satisfying the visual inspection acceptance criteria, freedom of motion of mechanical snubbers shall be verified using one of the following: (i) manually induced snubber movement; (ii) evaluation of in-place snubber piston setting; (iii) stroking the mechanical snubber through its full range of travel.

ATTACHMENT D

PROPOSED TECHNICAL SPECIFICATION 3/4.7.6

SAN ONOFRE NUCLEAR GENERATING STATION

UNIT 3

SURVEILLANCE REQUIREMENTS (Continued)

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to (a) the component or pipe and (b) the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.6.e or 4.7.6.f, as applicable. However, when a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

d. Functional Tests*

At least once per 18 months during shutdown, a representative sample of at least 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber of a type of that does not meet the functional test acceptance criteria of Specification 4.7.6.e or 4.7.6.f, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle

2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine motor, etc.)

3. Snubbers within 10 feet of the discharge from safety relief valve

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

SURVEILLANCE REQUIREMENTS (Continued)

i. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

j. Transient Event Inspections

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of safety systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following a determination that such an event has occurred. In addition to satisfying the visual inspection acceptance criteria, freedom of motion of mechanical snubbers shall be verified using one of the following: (i) manually induced snubber movement; (ii) evaluation of in-place snubber piston setting; (iii) stroking the mechanical snubber through its full range of travel.

DESCRIPTION OF PROPOSED CHANGES NPF-10/15-213 AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.1.1.3, "Moderator Temperature Coefficient".

Existing Technical Specifications

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

Proposed Technical Specifications

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

Description

The proposed change revises Technical Specification 3/4.1.1.3, "Moderator Temperature Coefficient". Technical Specification 3/4.1.1.3 defines limitations on moderator temperature coefficient (MTC) to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are performed to confirm the MTC value since this coefficient changes slowly due principally to the reduction in reactor coolant system (RCS) boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

Technical Specification 3/4.1.1.3 currently states that the moderator temperature coefficient shall be less negative than -2.5×10^{-4} delta $k/k/^{o}F$ at rated thermal power. The proposed change will state that the moderator temperature coefficient shall be less negative than -3.0×10^{-4} delta $k/k/^{o}F$. This change is required to mitigate the effect of double counting Control Element Assembly (CEA) rod worths in the Cycle 2 transient analyses. The resultant reactivity gain can therefore be applied to support the end-of-cycle operation with an anticipated more negative MTC value bounded by the proposed change.

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 proposed negative moderator temperature coefficient change was incorporated as a trade-off resulting from a new reactivity balancing which provides additional margin to the Cycle 2 transient analysis. There are no changes to the results of all transient analyses because of the proposed change. Since the existing safety analysis results are clearly within all acceptable criteria with respect to the system or component of concern as specified in the Standard Review Plan, Section 4.3, the proposed change thus remains accounted for in an equally conservative manner as before. The events most affected by the change are those characterized by a decrease in primary temperature. A detailed review of the most limiting transient event affected by the proposed change at the end of Cycle 2 shows that it will not impose any adverse impact nor result in any increase in the 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

   No change to operating procedures is involved. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident 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

   The proposed negative moderator temperature coefficient change was incorporated as a trade-off resulting from a new reactivity balancing which provides additional margin to the Cycle 2 transient analysis. The events most affected by the change are those characterized by a decrease in primary temperature and are bounded by the analyses presented in the Reload Analyses Report for Cycle 2. These analyses have already demonstrated that there will not be any increase in the consequences of an accident and the results of the change are clearly within all acceptable criteria with respect to the system or component of concern as specified in the Standard Review Plan, Section 4.3. Since there are no changes to the results of all transient analyses because of the proposed change, the proposed change thus remains accounted for in equally conservative manner as before. Consequently, the proposed change will not involve any reduction in safety margins.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a refinement of a previously used calculational model or design method. The proposed change is similar to Example (vi) in that the technical specification on the moderator temperature coefficient will reflect a relaxation of an assumption used in the Cycle 2 transient analysis. This revision is a trade-off of reactivity used in the analysis of the most limiting transient at the end of Cycle 2 without affecting any assumptions previously evaluated. Specifically, the proposed change pertains to a revision of uncertainty analyses relating to reactivity coefficients performed in accordance with the guidelines of the Standard Review Plan (SRP), Section 4.3, "Nuclear Design." This revision accounts for a reactivity gain mistakenly discredited as a result of double counting CEA rod worths in the existing safety analyses reported in the Cycle 2 Reload Analysis Report. The proposed change in the MTC value offsets this reactivity gain pursuant to the SRP, Section 4.3.3, so as to conserve the magnitude of overall uncertainties used in the safety analyses. Consequently, this change does not make changes in analytical methods or results of analyses previously found to be acceptable by the NRC and used to demonstrate conformance with the regulations. Furthermore, an evaluation of the most limiting transient shows that the present safety analyses remain valid and bounding. Thus, the proposed change to relax the Technical Specification MTC limit is compensated by an over-conservatism in CEA rod worths without changing the overall conclusion of the present safety analyses. It results in a refinement of uncertainties previously used in the safety analyses in accordance with the SRP, Section 4.3, and is therefore similar to Example (vi) of 48 FR 14870.

Safety and Significant Hazards Determination

Based on the above Safety Analysis, it is concluded that: (1) there is a 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.

SPW:5465F

ATTACHMENT A

Existing Technical Specifications, Unit 2

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, or

Less positive than 0.0 delta k/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

b. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

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

#See Special Test Exception 3.10.2.

ATTACHMENT B

Proposed Technical Specifications, Unit 2

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, or

Less positive than 0.0 delta k/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

b. Less negative than -3.0×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

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

#See Special Test Exception 3.10.2.

ATTACHMENT C

Existing Technical Specifications, Unit 3

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, or less positive than 0.0 delta k/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

b. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

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

#See Special Test Exception 3.10.2.

ATTACHMENT D

Proposed Technical Specifications, Unit 3

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, or less positive than 0.0 delta k/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

b. Less negative than -3.0×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

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

#See Special Test Exception 3.10.2.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-214

This is a request to revise Technical Specification 3/4.4.7 "Specific Activity" and Technical Specification 6.9.1.5 "Annual Reports."

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 proposed change revises Technical Specification 3/4.4.7, "Specific Activity," and Technical Specification 6.9.1.5, "Annual Reports." Technical Specification 3/4.4.7 defines allowable limits for concentrations of radioactive isotopes in the reactor coolant system (RCS), specifies a sampling and analysis program to verify RCS activity is within the limits, and defines actions to be taken in the event that RCS activity exceeds the specified limits. When the specified limits are exceeded, Technical Specification 3/4.4.7 allows continued operation for up to 48 continuous hours provided that RCS activity remains within the region of acceptable operation defined by Figure 3.4-1 and provided that the cumulative operating time does not exceed 800 hours in any consecutive 12-month period. In addition, a special report is required if 500 consecutive hours are exceeded in any consecutive six-month period. If the specific activity exceeds the specified limits for more than 48 consecutive hours, a plant shutdown would be required within the next six hours. The actions also require submittal of a License Event Report (LER) within the next 30 days. The LER is to include: 1) reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; 2) fuel burnup by core region; 3) cleanup flow history starting 48 hours prior to the first sample in which the limit was exceeded; 4) history of de-gassing, if any, starting 48 hours prior to the first sample in which the limit was exceeded; and 5) the time duration when the RCS specific activity exceeded one microcurie per gram dose equivalent I-131.

The proposed change would revise existing action requirements to delete the 800 hour per year consecutive limit on operation while exceeding the specific activity limit, eliminate the special reporting requirement when 500 hours are exceeded, and remove the requirement for an LER to be submitted within 30 days. Instead of requiring a License Event Report, the proposed change would revise Technical Specification 6.9.1.5 to include the currently required information in the annual report.

The proposed change also removes redundancy between the existing actions and surveillance requirements. In addition to specifying the reporting requirements, the action also specifies performance with the surveillance requirement sampling and analysis program. Performance of the surveillance is required regardless of being in the action or not. Therefore, the proposed change deletes this redundancy from the action.

Safety Analysis

The proposed change described above shall be deemed to involve significant hazards considerations 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 reduced reporting requirements for primary coolant activity will not impact safe operation or the consequences of an accident previously evaluated.

   Technical Specification 3/4.4.7 prohibits continued plant operation if coolant iodine activity limits are exceeded for 800 hours in a 12-month period. This requirement reduces the probability of high iodine being present in the RCS coincident with a postulated steam generator tube rupture, thereby reducing the potential offsite dose consequences. Elimination of this requirement is acceptable because the improvement in the quality of fuel has reduced the potential for operation with high coolant iodine activity to the point where the 800 hour limit would not likely be approached. In addition, 10CFR50.72 requires immediate NRC notification of fuel cladding failures that exceed the expected value or that are caused by unexpected factors. Therefore, this Technical Specification limit is no longer considered necessary on the basis that proper fuel management at San Onofre Units 2 and 3 and existing reporting requirements should preclude ever approaching the limit.

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

   Response: No

   The proposed change does not reduce surveillance of primary coolant iodine activity or preclude responsible actions to maintain low primary coolant iodine activity. Therefore, the primary coolant activity levels will not approach the accumulated time limit and result in a new or different kind of accident that has not been 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

   The margin of safety for Technical Specification 3/4.4.7 is established by the limits of primary coolant activity in 3/4.4.7. The proposed change does not change the limits on primary coolant activity levels during operation. With improved fuel quality, the cumulative operating time with high iodine activity should not approach the 800 hours limit. Therefore, operation of the facility in accordance with the proposed change will not involve a significant reduction in a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (i) relates to a purely administrative change to technical specifications: for example, a change to achieve consistency throughout technical specifications, correction of an error, or a change in nomenclature. Example (vii) relates to a change to make a license conform to changes in the regulations where the license change results in very minor changes to facility operations currently in keeping with the regulations.

In this case, 10CFR50.34 requires technical specifications covering a number of diverse aspects of facility operation. Conformance with the standard technical specifications provides an acceptable means of meeting the requirements of 10CFR50.34.

NRC Generic Letter 85-19 dated September 27, 1985 revised the standard technical specifications relating to specific activity. Generic Letter 85-19 incorporates the above proposed change into the standard technical specifications. This change will have a minor impact on facility operation since it only affects reporting requirements and actions to be taken when specific activity limits are exceeded. The specific activity limits are not revised by the proposed change. Because the proposed change has only a minor affect on facility operation and brings the technical specifications in conformance with the standard technical specification, as revised by Generic Letter 85-19, the proposed change is similar to example (vii). The proposed change would eliminate redundancy between the existing action and surveillance requirements. This change is editorial and does not change existing requirements to perform sampling and analysis in accordance with the surveillance; therefore, this change is similar to example (i). Because the proposed changes are similar to examples (i) and (vii), they do not involve significant hazards considerations.

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 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 A

Existing Technical Specifications, Unit 2

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microcuries/gram.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

c. With the specific activity of the primary coolant greater than 100/$\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

*With T_{avg} greater than or equal to 500°F.

ACTION: (Continued)

MODES 1, 2, 3, 4 and 5:

c. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the San Onofre site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated manrem exposure according to work and job functions,** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

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* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

** This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ATTACHMENT B

Proposed Technical Specifications, Unit 2

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram.

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

ACTION:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the San Onofre site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

(6.9.1.5 Cont'd)

Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.7. The following information shall be included in these reports: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ATTACHMENT C

Existing Technical Specifications, Unit 3

REACTOR COOLANT SYSTEM

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

c. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

*With T_{avg} greater than or equal to 500°F.

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ACTION: (Continued)

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

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CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the San Onofre site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

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SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermally induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermally induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

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STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated manrem exposure according to work and job functions,** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

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* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

** This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

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ATTACHMENT D

Proposed Technical Specifications, Unit 3

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram.

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

ACTION:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the San Onofre site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

(6.9.1.5 Cont'd)

Reports required on an annual basis shall include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.7. The following information shall be included in these reports: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last iostopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

DESCRIPTION OF PROPOSED CHANGES
NPF-10/15-210 REVISION 2
AND SAFETY ANALYSIS

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

Existing Specifications

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

Proposed Specifications

Units 2 and 3: See Attachment C

Description

The proposed change would delete, in its entirety Technical Specification 3/4.6.2.2 "Iodine Removal System," and replace it with a new technical specification requiring trisodium phosphate in the containment emergency sump area.

Technical Specification 3/4.6.2.2, "Iodine Removal System" requires that a spray additive tank, containing at least 1456 gallons of between 40 and 44% by weight of NaOH solution, and two chemical addition pumps be operable in Modes 1, 2 and 3. The original purpose of this Iodine Removal System was to ensure that in the event of a LOCA a sufficient amount of NaOH will be added to the containment spray to raise the pH to between 8 and 9 during the initial phase of the spray. The effects of the increased pH levels are to increase the iodine removal capability of the spray and the iodine retention in the sump.

An additional function of the NaOH in the Iodine Removal System, during the long term recirculation phase, is to maintain the pH level of sump at $\geq$ 7.0 to minimize the potential for chloride induced stress corrosion cracking of austenitic stainless steel.

A new analysis utilizing recent changes in NRC methodology (NUREG-0800, Section 6.5.2, Rev. 1), combined with knowledge gained from recent studies on the behavior of iodine in the post-LOCA environment, demonstrates that the deletion of the Spray Additive Tank does not significantly change the calculated offsite thyroid doses. The pH of the containment spray does not need to be increased during the initial phase of containment spray during a LOCA to enhance iodine removal.

However, in the post-LOCA recirculation phase, the Emergency Core Cooling System (ECCS) solution pH must be increased to $\geq$ 7.0 to minimize chloride induced stress corrosion cracking of austenitic stainless steel components, and to minimize the hydrogen produced by the corrosion of galvanized surfaces and zinc based paints. To accomplish this increase in the ECCS solution pH, a new Technical Specification is proposed to replace Technical Specification 3.6.2.2. This new Technical Specification requires the presence of a specified amount of trisodium phosphate in the containment area. This amount of trisodium phosphate will maintain long term pH control in the ECCS recirculation solution, thereby minimizing the potential for chloride induced stress corrosion and maximizing iodine retention in the sump solution.

Safety Analysis

The proposed changes 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 this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The plant systems, in which a change is proposed, are intended to respond to and mitigate the effects of a LOCA. The proposed changes have no effect on the probability of the occurrence of a LOCA.

   A new analysis utilizing recent changes in the Standard Review Plan and knowledge gained from recent studies on the behavior of iodine in the post-LOCA environment has shown that the deletion of the Iodine Removal System, and its replacement with a sump pH control system will not significantly affect the radiological consequences of a postulated LOCA and the calculated doses will remain well within the 10 CFR 100 guidelines. In addition, the use of TSP for long term recirculation phase pH control meets all the requirements for control of chloride stress corrosion and maximizes iodine retention in the sump solution.

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 substitutes a passive system for the active system currently used to mitigate the consequences of an accident. It does not affect any system involved in the initiation of an accident. Therefore, the proposed 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 reduction in a margin of safety.

   Response: No

The radiological consequences of a postulated LOCA will not increase relative to the 10 CFR 100 guidelines, nor will the probability of chloride induced stress corrosion cracking increase.

The Commission has provided guidance for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards consideration. Example (vi) relates to a change which either may result in some increase in the probability or consequences of a previously-analyzed accident or may in some way reduce a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP).

SRP Sections 6.1.1, "Engineered Safety Features Materials," 6.5.2, "Containment Spray as a Fission Product Cleanup System," and 15.6.5, "Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant System," define the pertinent acceptance criteria. SRP Section 6.1.1 requires that the composition of containment spray and core cooling water be controlled to ensure a minimum pH of 7.0 following a loss of coolant accident to inhibit initiation of stress corrosion cracking. SRP Section 6.5.2 defines conditions under which the containment spray system can be credited for fission product removal. SRP Section 15.6.5 defines, by reference to 10 CFR 100, the post accident dose limits.

The only impact that the proposed Technical Specification change has on this system is the deletion of the use of NaOH in the initial containment spray phase following a postulated LOCA, and the substitution of trisodium phosphate for NaOH in the sump solution during the long term recirculation phase. Consistent with SRP Section 6.5.2, no credit is taken for containment spray removal of elemental iodine. The current analysis taking credit for NaOH addition calculates a 0-2 hour Exclusion Area Boundary (EAB) thyroid dose of 86.0 rem and a 0-30 day Low Population Zone (LPZ) thyroid dose of 11.5 rem. The 10 CFR 100 acceptance criteria are 300 rem for both categories. For the new analysis, the corresponding conservative case EAB thyroid dose was 76.2 rem and LPZ thyroid dose of 12.2 rem. An optimized case resulted in calculated EAB thyroid dose of 57.7 rem and LPZ thyroid dose of 8.7 rem. Depending on the degree of conservatism in the new analysis, the deletion of the Spray Additive Tank may slightly increase or decrease the calculated thyroid dose at the LPZ, and will in all cases reduce the thyroid dose at the Exclusion Area Boundary. It should be noted that in all cases there is significant margin between the calculated thyroid doses and the limits defined in 10 CFR 100, and this margin is essentially independent of whether the Spray Additive Tank is operable, or if the SAT is deleted and the Sump pH Control System is operable. Thus, the proposed change meets the dose acceptance criteria of SRP Section 15.6.5.

The proposed requirements will assure a post accident sump pH ≥ 7.0 meeting the SRP 6.1.1 requirements to minimize the potential for chloride stress corrosion, the generation of hydrogen or the environmental qualification of equipment. Therefore, because the proposed change meets the SRP acceptance criteria, it is similar to example (vi).

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.

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ATTACHMENT A

UNIT 2 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.

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

UNIT 3 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 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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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 C

UNITS 2 AND 3 PROPOSED SPECIFICATION

CONTAINMENT SYSTEMS

RECIRCULATION FLOW PH CONTROL

LIMITING CONDITION FOR OPERATION

3.6.2.2 The recirculation flow pH control system shall be operable with a minimum of 15,400 lbs. (256 cu. ft.) of trisodium phosphate (w/12 hydrates), or equivalent, available in the storage racks in the containment.

APPLICABILITY: Modes 1, 2 and 3

ACTION:

With less than the required amount of trisodium phosphate available, restore the system to the correct amount within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The recirculation flow pH control system shall be demonstrated operable during each refueling outage by:

a. Visually verifying that the TSP storage racks have maintained their integrity and the TSP containers contain a minimum of 15,400 lbs. (256 cu. ft.) of TSP (w/12 hydrates) or equivalent.

b. Verifying that when a sample of less than 3.03 grams of trisodium phosphate (w/12 hydrates) or equivalent, selected at random from one of the storage racks inside of containment, is submerged, without agitation, in at least 1 litre of 120 ± 10 degrees-F borated demineralized water borated to at least 2482 ppm boron, allowed to stand for 4 hours, then decanted and mixed, the pH of the solution is greater than or equal to 7.0.