DESCRIPTION OF PROPOSED CHANGE NPF-10-85 AND NPF-15-85
AND SAFETY ANALYSIS

This is a request to revise Technical Specification Table 3.3-2, Reactor Protective Instrumentation Response Times.

<u>Existing Specifications</u>

<u>Unit 2</u>

See Attachment A

<u>Unit 3</u>

See Attachment C

<u>Proposed Specifications</u>

<u>Unit 2</u>

See Attachment B

<u>Unit 3</u>

See Attachment D

<u>Description</u>

The effect of degraded RTD response times on the reactor protection system has been evaluated by analyzing events which induce significant primary coolant temperature transients. The objective of these analyses was to determine the magnitude of any change in system response resulting from degraded RTD's, and to specify penalty factors for the CPC's and COLSS which are sufficient to compensate for this degradation. Consequently, application of the resulting penalty factors will ensure that DNBR and LPD trip functions are unimpaired by degraded RTD response characteristics.

The basis for the proposed technical specification change has been to determine penalty factors which may be applied via CPC addressable constants, or penalties to the LCO Power Operating Limit that will ensure that the CPC trip functions will not be compromised if the input RTD response times degrade beyond the value assumed in the current software of the CPC.

<u>Safety Evaluation</u>

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

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

Response: No

The CPC responds to design basis transient events based in part on information from RTD's. These proposed changes specify CPC penalty factors to compensate for degraded RTD response times. This allows the CPC to continue to provide the same degree of protection for accidents previously evaluated in the FSAR. Also, because the CPC responds to design basis events, these changes will not increase the probability of accidents previously evaluated.

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

Response: No

These proposed changes specify CPC penalty factors to compensate for degraded RTD response time. Because the CPC responds to design basis events, these changes do not create the possibility for a new or different kind of accident from any accident previously evaluated.

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

Response: No

The events analyzed to evaluate the effect of degraded RTD response times were:

Loss of Load
Single CEA Withdrawal
CEA Bank Withdrawal
Excess Load
Single CEA Drop
Loss of Feedwater
Instantaneous Closure of a Single MSIV

The Loss of Load event was analyzed to determine the effect of degraded RTD response characteristics on the core inlet temperature used by the CPCS. At the limiting conditions, the analyses incorporating RTD response times of 8.0, 10.0, and 13.0 seconds showed that power penalties in Table 1.1 will ensure conservative CPC DNBR and local power density (LPD) calculations.

The single CEA Withdrawal event from 1% power was analyzed to determine the effect on the dynamic term of the thermal power calculation performed by the CPCS. For single rod deviation events, neutron flux power is not credited in the analysis to determine CPC response. Since the thermal power calculation was based on the rise in coolant enthalpy across the core and the dynamic thermal power

calculation uses the hot leg temperature, changes in RTD response times significantly affect this portion of the CPCS calculations. At the limiting conditions, the analyses incorporating RTD response times of 8.0, 10.0, and 13.0 seconds showed that the power penalties of Table 1.1 will ensure conservative CPC DNBR and LPD calculations.

The Excess Load event was analyzed to determine the effect of degraded RTD response characteristics on the core inlet temperature used by the CPCS. The effect on CPC is an under-prediction of core power estimates that is partially compensated by an over-prediction of the inlet temperature used in the DNB calculations. At the limiting condition, the analyses using RTD response times of 8.0, 10.0, and 13.0 seconds showed that the power penalties of Table 1.1 will ensure conservative CPC DNBR and LPD calculations.

It was determined that the penalties resulting from these analyses are sufficient to accommodate any non-conservatisms in CPC protection for CEA Bank Withdrawal, Single CEA Drop, and Loss of Feedwater events.

Asymmetric Steam Generator Transient Protection is provided by the CPCS using input from the cold leg RTDs. The limiting Asymmetric Steam Generator Transient event was reanalyzed for degraded RTD time constants. The asymmetric steam generator trip function monitors a dynamically compensated temperature difference between cold legs, and will initiate a reactor trip when the temperature difference exceeds 180°F. An analysis of the Instantaneous Closure of a Single MSIV event was performed to determine the additional DNBR required overpower margin (ROPM) needed to ensure that the fixed CPC asymmetric steam generator trip setpoint provides adequate protection. The additional ROPM for 8.0, 10.0, and 13.0 second RTD time constants was determined to be 4.0%, 5.0%, and 7.0% of full power, respectively. This additional ROPM can be maintained either by applying the penalties to the power operating limit in the Core Operating Limit Supervisory System (COLSS) or, with COLSS out of service, by applying the penalties to the affected CPC channel(s) being used for monitoring the DNBR LCO using existing Technical Specification Figure 3.2-2.

Application of the CPC and power operating limit penalties in the proposed technical specification allows the protective features of the CPCS to be maintained uncompromised in the event that CPC input RTD response times are degraded. Consequently, the margin of safety provided by the CPCS trips is clerly unchanged by this proposed technical specification. In addition, other PPS trip functions such as high pressurizer pressure, low steam generator water level, and low steam generator pressure complement the CPC DNBR and LPD trips for the events where RTD response time degradation affects the CPC trips.

The application if increased CPC penalty factors to compensate for RTD response time degradation will allow continued plant operation and could preclude the need for a shutdown to repair the degraded RTDs.

The application of the CPC and power operating limit penalties in the proposed changes allow the protective features of the CPC to be maintained in the event that CPC RTD input response times are degraded. Since the CPC will continue to provide the same degree of protection for the FSAR design basis events, before as well as after the proposed change, the margin of safety provided by the CPC is unchanged.

Safety and Significant Hazards Determination

Based on the above discussion, Proposed Changes NPF-10-85 and NPF-15-85 do not involve a significant hazard consideration in that they do not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In addition, 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.

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TABLE 1.1

CPC PENALTY FACTOR INCREASES FOR RTD DEGRADATION

	DNB AT POWER (BERR0 INCREASE)			DNB FLUX POWER (BERR2 INCREASE)			LOCAL POWER DENSITY (BERR4 INCREASE)		
RTD TIME CONSTANT	8.0 sec	10.0 sec	13.0 sec	8.0 sec	10.0 sec	13.0 sec	8.0 sec	10.0 sec	13.0 sec
Event									
Loss of Load	(1)	2.0	4.5	2.5	4.0	5.5	(1)	(1)	(1)
Excess of Load	(1)	0.5	1.0	3.5	4.0	4.5	(1)	(1)	(1)
Single CEAW	(1)	3.5	10.5	(2)	(2)	(2)	3.0	9.0	17.0
Required CPC Penalty Factor Change for COLSS in Service	0	3.5	10.5	3.5	4.0	5.5	3.0	9.0	17.0
Required CPC Penalty Factor Change for COLSS Not in Service(3)	0	3.5	10.5	4.0(4)	5.0(4)	7.0(4)	3.0	9.0	17.0

(1) No additional penalty required for this event.

(2) This event not applicable.

(3) Penalty to be applied only to the CPC channel(s) with degraded RTD's.

(4) Values based on the Asymmetric Steam Generator event.

ATTACHMENT A

(Existing Specification)

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1 or 3.2-2, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

TABLE 3.3-[illegible] (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 6.0 seconds when the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

ATTACHMENT B

(Proposed Specification)

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1 or 3.2-2, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2 and that the conditions of Table 3.3-2b are satisfied.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

TABLE 3.3-[illegible] (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to13.0 seconds when the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature. Adjustments to the CPC addressable constants in Table 3.3-2a and reductions in the DNBR Power Operating Limit in Table 3.3-2b shall be made to accommodate measured values of the RTD time constants.

TABLE 3.3-2a

INCREASES IN BERRO, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

RTD DELAY TIME τ	BERRO INCREASE %	BERR2 INCREASE %	BERR4 INCREASE %
$\tau \leq 6.0$ sec	0.0	0.0	0.0
6.0 sec $< \tau \leq$ 8.0 sec	0.0	3.5	3.0
8.0 sec $< \tau \leq$ 10.0 sec	3.5	4.0	9.0
10.0 sec $< \tau \leq$ 13.0 sec	10.5	5.5	17.0

NOTE: BERR term increases are not cumulative, i.e., if the values of the BERR terms are currently 10.0, then for an RTD delay time of >6.0 to ≤8.0 sec, BERRO = 10.0 + 0.0 = 10.0, BERR2 = 10.0 + 3.5 = 13.5, and BERR4 = 10.0 + 3.0 = 13.0. Computed values in this paragraph and below are examples only.

For RTD delay times >8.0 to ≤10.0 sec, BERRO = 10.0 + 3.5 = 13.5, BERR2 = 10.0 + 4.0 = 14.0, and BERR4 = 10.0 + 9.0 = 19.0.

Increases are similarly applied for RTD delay times >10.0 to ≤13.0 sec.

NOTE: When any of the above increases are applied to the BERR terms for any CPC channel, the COLSS constant EPOL 2 is reduced by 0.04.

TABLE 3.3-2b

DNBR LCO POWER OPERATING LIMIT ADJUSTMENTS

RTD Delay Time (sec)	Adjustment to EPOL1[1], COLSS In Service (% power)	Adjustment to BERR2[1, 2], COLSS Out-of-Service (% power)
$\tau \leq 6.0$ sec	0.0	0.0
6.0 sec $< \tau \leq$ 8.0 sec	-4.0	+4.0
8.0 sec $< \tau \leq$ 10.0 sec	-5.0	+5.0
10.0 sec $< \tau \leq$ 13.0 sec	-7.0	+7.0

NOTES:

1. Adjustments are not cumulative; i.e., if τ increases from 7.0 seconds to 9.0 seconds, EPOL1 is reduced by 5.0 from its original value, not 4.0 + 5.0 = 9.0 from its original value.

2. If COLSS is out-of-service, these adjustments are to be used in place of, not in addition to, the increases required by Table 3.3-2a, and the limit in Figure 3.2-2 must be maintained for all operable CPC channels.

ATTACHMENT C

(Existing Specifications)

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1 or 3.2-2, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

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TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

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*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 6.0 seconds when the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

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ATTACHMENT D

(Proposed Specifications)

POWER DISTRIBUTION LIMITS

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1 or 3.2-2, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2 and that the conditions of Table 3.3-2b are satisfied.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

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*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 13.0 seconds when the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature. Adjustments to the CPC addressable constants in Table 3.3-2a and reductions in the DNBR Power Operating Limit in Table 3.3-2b shall be made to accommodate measured values of RTD time constants.

TABLE 3.3-2a

INCREASES IN BERR0, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

RTD DELAY TIME τ	BERR0 INCREASE %	BERR2 INCREASE %	BERR4 INCREASE %
$\tau \leq 6.0$ sec	0.0	0.0	0.0
6.0 sec $< \tau \leq$ 8.0 sec	0.0	3.5	3.0
8.0 sec $< \tau \leq$ 10.0 sec	3.5	4.0	9.0
10.0 sec $< \tau \leq$ 13.0 sec	10.5	5.5	17.0

NOTE: BERR term increases are not cumulative, i.e., if the values of the BERR terms are currently 10.0, then for an RTD delay time of >6.0 to ≤8.0 sec, BERR0 = 10.0 + 0.0 = 10.0, BERR2 = 10.0 + 3.5 = 13.5, and BERR4 = 10.0 + 3.0 = 13.0. Computed values in this paragraph and below are examples only.

For RTD delay times >8.0 to ≤10.0 sec, BERR0 = 10.0 + 3.5 = 13.5, BERR2 = 10.0 + 4.0 = 14.0, and BERR4 = 10.0 + 9.0 = 19.0.

Increases are similarly applied for RTD delay times >10.0 to ≤13.0 sec.

NOTE: When any of the above increases are applied to the BERR terms for any CPC channel, the COLSS constant EPOL2 is reduced by 0.04.

TABLE 3.3-2b

DNBR LCO POWER OPERATING LIMIT ADJUSTMENTS

RTD Delay Time (sec)	Adjustment to EPOL1[1], COLSS In Service (% power)	Adjustment to BERR2[1, 2], COLSS Out-of-Service (% power)
$\tau \leq 6.0$ sec	0.0	0.0
6.0 sec $< \tau \leq$ 8.0 sec	-4.0	+4.0
8.0 sec $< \tau \leq$ 10.0 sec	-5.0	+5.0
10.0 sec $< \tau \leq$ 13.0 sec	-7.0	+7.0

NOTES:

1. Adjustments are not cumulative; i.e., if τ increases from 7.0 seconds to 9.0 seconds, EPOL1 is reduced by 5.0 from its original value, not 4.0 + 5.0 = 9.0 from its original value.

2. If COLSS is out-of-service, these adjustments are to be used in place of, not in addition to, the increases required by Table 3.3-2a and the limit in Figure 3.2-2 must be maintained for all operable CPC channels.

DESCRIPTION OF PROPOSED CHANGE NPF-10-89 AND NPF-15-89 AND SAFETY EVALUATION

This is a request to revise Technical Specification 3/4.4.4, Steam Generators.

Existing Technical Specification

See Attachment A. The existing specification is identical for Units 2 and 3.

Proposed Technical Specification

See Attachment B. The proposed specification is identical for Units 2 and 3.

Reason for Proposed Change

The purpose of this proposed change is to clarify and simplify the wording of the existing Technical Specification. In cases where more than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective, the inspection frequency in the proposed change will be increased to a 12 to 24 month interval. This change allows greater flexibility in scheduling subsequent S/G inspections. The requirements discussed in the bases are still applicable.

A summary of each change is provided as follows:

Technical Specification Change	Change Summary
3.4.4 ACTION (old)	"Increasing Tavg above 200°F" revised to "Entry into MODES 1, 2, 3 or 4" since the use of operational modes is clearer than using average coolant temperature.
4.4.4.0 (old)	Revised wording incorporated into 4.4.4.1 (new).
4.4.4.1 (old)	The operability statement has been included in 4.4.4.6 (new). The reference to Table 4.4-1 has been deleted. Table 4.4-1 is now included in 4.4.4.2 (new).
4.4.4.2 (old)	The actions specified in Table 4.4-2 as stated in the first sentence are now in 4.4.4.6 (new). The inspection frequency and acceptance criteria as stated in the second sentence are now in 4.4.4.1 (new) and Table 4.4-2 (new), respectively. The total number of tubes requiring inspection as stated in the first part of the third sentence is now in 4.4.4.2 (new). The random basis selection requirement as stated in the last part of the third sentence is now in 4.4.4.3 (new).

4.4.4.2.a (old)	This similar plant experience requirement has been restated in 4.4.4.3.b (new).
4.4.4.2.b (old)	The first sample requirements have been restated in 4.4.4.3 (new), 4.4.4.4 (new) and 4.4.4.5 (new).
4.4.4.2.b.1 (old)	Renumbered as 4.4.4.3.a (new).
4.4.4.2.b.2 (old)	This experience requirement has been restated in 4.4.4.3.b (new).
4.4.4.2.b.3 (old)	The inspection method requirement in the first sentence has been restated in 4.4.4.4 (new). The blocked tube requirements in the second sentence have been renumbered as 4.4.4.5 (new).
4.4.4.2.c (old)	The criteria for the second and third samples are restated in 4.4.4.4 (new).
4.4.4.2 (old)	The category definitions have been restated in Table 4.4-2 (new).
4.4.4.3.a (old)	The inspection frequency requirements in the first two sentences have been restated in 4.4.4.1.a (new). The inspection frequency requirements in the third sentence have been restated in 4.4.4.1.b (new).
4.4.4.3.b (old)	The inspection frequency requirements have been restated in 4.4.4.1.b (new) with the following exception. If the inspection results fall into Category C-3, the inspection frequency in the proposed change will be increased to a 12 to 24 month interval. This change allows greater flexibility in scheduling subsequent inspections.
4.4.4.3.c (old)	These additional inspection requirements have been restated in 4.4.4.1.c (new).
4.4.4.4.a.1 (old)	The definition of imperfection has been restated in Note 4 to Table 4.4-2 (new).
4.4.4.4.a.2 (old)	The definition of degradation has been restated in Note 2 to Table 4.4-2 (new).
4.4.4.4.a.3 (old)	The definition of degraded tube has been restated in Note 2 to Table 4.4-2 (new).
4.4.4.4.a.4 (old)	The definition of percent degradation has been deleted since it was not referenced in the old T.S.
4.4.4.4.a.5 (old)	The definition of defect has been restated in Note 3 to Table 4.4-2 (new).

4.4.4.4.a.6 (old)	The definition of plugging limit has been restated in Note 3 to Table 4.4-2 (new).
4.4.4.4.a.7 (old)	The definition of unserviceable has been deleted since it was not referenced in the old T.S.
4.4.4.4.a.8 (old)	The definition of tube inspection has been included in 4.4.4.4 (new).
4.4.4.4.a.9 (old)	The definition of preservice inspection has been deleted since Units 2 and 3 are operating plants.
4.4.4.4.b (old)	The definition of OPERABLE has been restated in 4.4.4.6 (new).
4.4.4.5.a (old)	The special reporting requirement on plugged tubes has been restated in 4.4.4.7.a (new) with the following exception. The time to submit the report has been changed from 15 to 30 days to allow sufficient time to review preliminary inspection data.
4.4.4.5.b (old)	The special reporting requirement on the inspection results has been restated in 4.4.4.7.b (new).
4.4.4.5.c (old)	The prompt reporting requirement if the results fall into Category C-3 has been restated in 4.4.4.7.c (new).
Table 4.4-1 (old)	The inspection requirements have been restated in 4.4.4.2 (new).
Table 4.4-2 (old)	Table 4.4-2 has been simplified for the SONGS 2/3 design in Table 4.4-2 (new).
Figure 4.4-1 (old)	The tube wall thinning acceptance criteria has not changed.

Safety Evaluation

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

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

   Response: No

   The steam generator tube inspection program as previously reviewed by the NRC staff in the February 1981 Safety Evaluation Report for Units 2 and 3 has not changed in content or scope. The program is in compliance with the guidelines of Regulatory Guide 1.83, Revision 1 and meets the essential requirements of NUREG-0212. The program also complies with the inspection requirements of Section XI of the ASME code.

The proposed change clarifies and simplifies the present wording and increases the flexibility of inspection frequencies as allowed by the ASME code.

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

   This proposed change is editorial in nature and has no safety or environmental significance.

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

   Response: No

   Section 4.4.4.1.b changes the subsequent inspection frequency if the results of a 40 month inspection falls into Category C-3. The current wording states the frequency shall be increased to at least once per 20 months as opposed to the proposed wording which states the frequency shall be increased to a 12 to 24 month interval. The proposed wording comes directly from ASME Chapter XI, Section IWB-2420. This section is incorporated by reference into Regulatory Guide 1.83, Revision 1, Sections C.6.b and c. Although this change may cause an increase in inspection intervals from 20 to 24 months it maximizes the inspection interval allowed under Regulatory Guide 1.83 and the ASME Code and thus is deemed not to involve a significant reduction in a margin of safety.

   The special reporting requirement following completion of the inspection has been changed from 15 to 30 days. This change brings the reporting requirement in line with other reporting requirements such as LER's and allows sufficient time to review preliminary inspection data.

The proposed clarification and editorial change is similar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983, in that it may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. The proposed change may be perceived to reduce a safety margin with respect to 20 versus 24 month inspection intervals. However, the change is within the acceptance criteria of Section 5.4.2.2 of the Standard Review Plan.

Safety and Significant Hazards Consideration Determination

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

ATTACHMENT A

3/4.4.4 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.4 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

4.4.4.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.4.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.4.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.4.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.4.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.4.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calender months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in both sets of inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.4.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

4.4.4.4 Acceptance Criteria

a. As used in this Specification

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

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

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

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

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

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to the nominal tube wall thickness as determined in Figure 4.4-1.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.4.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.4.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Table 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. Prompt notification to NRC pursuant to specification 6.9.1	All other S. G.s are C-1	None	N/A	N/A
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1	N/A	N/A

$S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

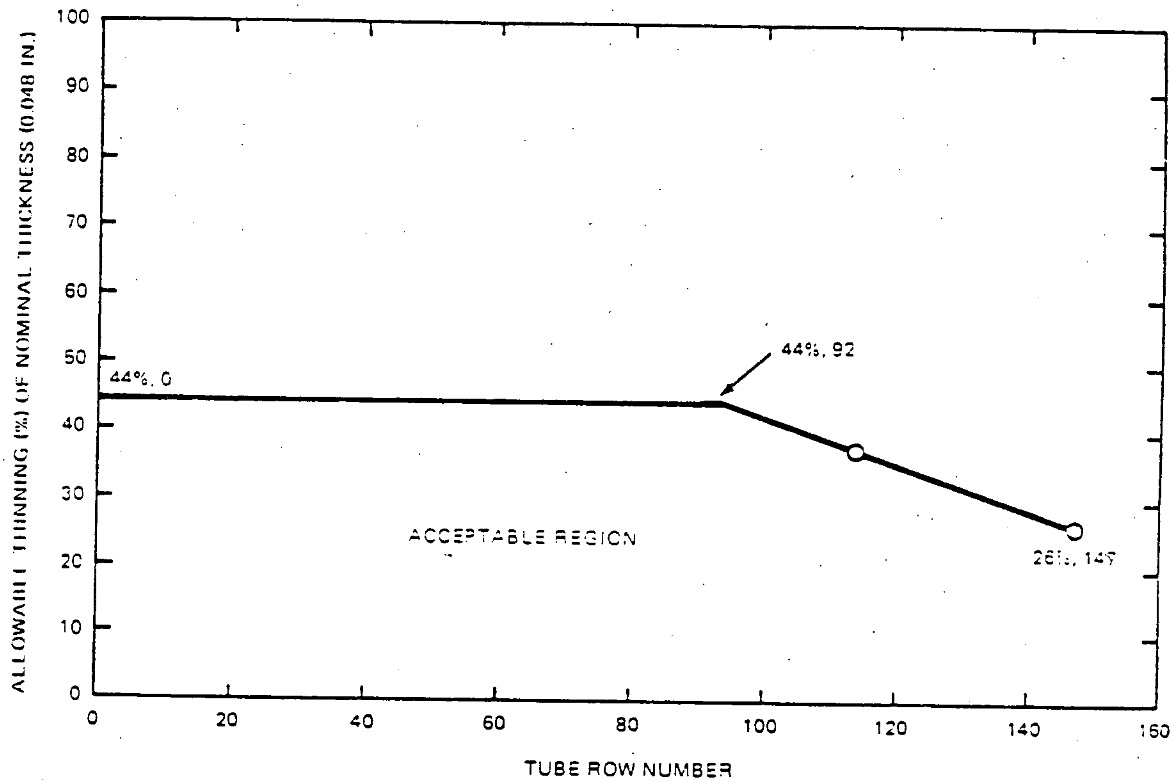

Figure 4.4-1
TUBE WALL THINNING ACCEPTANCE CRITERIA

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of Figure 4.4-1. Figure 4.4-1 was developed as a result of analyses performed for Primary Loop Pipe Break (PLPB) plus Design Basis Earthquake (DBE) and Main Steam Line Break (MSLB) plus DBE. These analyses examined families of tube rows as related to the number of vertical support grids. As horizontal tube spans become longer the loads become greater in spite of the increased number of tube supports. These results are controlling items in allowable tube wall thinning for tube rows 92 to 147. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

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ATTACHMENT B

REACTOR COOLANT SYSTEM

3/4.4.4 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.4 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to entry into MODES 1, 2, 3 or 4.

SURVEILLANCE REQUIREMENTS

4.4.4.1 Each steam generator shall be demonstrated OPERABLE by performance of inservice inspections of steam generator tubes at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection, except as provided in 4.4.4.1.b, below.

b. If two consecutive inservice inspections result in both sets of inspection results falling into the C-1 category, the inspection interval may be extended to a maximum of once per 40 months. However, if the results of the inservice inspection of a steam generator conducted at 40 month intervals fall into Category C-3, the inspection frequency shall be returned to an interval of not less than 12 nor more than 24 calendar months. When two subsequent consecutive inspection results fall into Category C-1 the interval may again be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

4.4.4.2 A minimum of 3% (561) of the 9350 tubes in each steam generator shall be inspected, except that the inservice inspection may be limited to one steam generator on a rotating schedule encompassing 6% (1122) of the tubes in that steam generator if the results of previous inspections indicate that both steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one steam generator may be found to be more severe than the other steam generator. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

4.4.4.3 The tubes selected for these inspections shall be selected on a random basis except the first sample of tubes selected for each inservice inspection of each steam generator shall include:

a. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

b. Tubes in those areas where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the minimum sample shall be from these critical areas.

4.4.4.4 Tubes in the first sample shall be inspected from the point of entry on the hot leg side, completely around the U-bend, to the top support of the cold leg. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

a. The tubes selected for these samples include the tubes from those areas of the tube sheet array where imperfections were previously found.

b. The inspections include those portions of the tubes where imperfections were previously found.

4.4.4.5 If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

4.4.4.6 The steam generator shall be designated as OPERABLE after completing the corresponding actions required by Table 4.4-2:

4.4.4.7 Reports

a. Within 30 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of a degraded tube.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 require prompt notification to the Commission pursuant to 10 CFR 50.72(b)(2)(i).

TABLE 4.4-2

STEAM GENERTOR TUBE INSPECTION

CORRECTIVE ACTION

Category	ADDITIONAL ACTION REQUIRED		
	1st Sample Inspection	2nd Sample Inspection	3rd Sample Inspection
C1	None	None	None
C2	Plug defective tubes and inspect at least an additional 2S tubes in this S/G	Plug defective tubes and inspect at least an additional 4S tubes in this S/G	Plug defective tubes
C3	Inspect all tubes in this S/G, plug defective tubes, and inspect (or expand the 1st sample to) 2S tubes in the other S/G	Inspect all tubes in this S/G and plug defective tubes	Inspect all tubes in this S/G and plug defective tubes

$S = \frac{1122}{n}$ Where n is the number of steam generators inspected

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	Between 5% and 10% of the total tubes inspected are degraded tubes, or one or more tubes, but not more than 1% of the total tubes inspected are defective.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Notes

1. In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

2. <u>Degraded Tube</u> means a tube containing service-induced cracking, wastage, wear, general corrosion or an imperfection occurring on either the inside or outside of a tube which is greater than or equal to 20% of the nominal wall thickness.

3. <u>Defective Tube</u> means a tube containing an imperfection of such severity that it exceeds the plugging limit shown in Figure 4.4-1.

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

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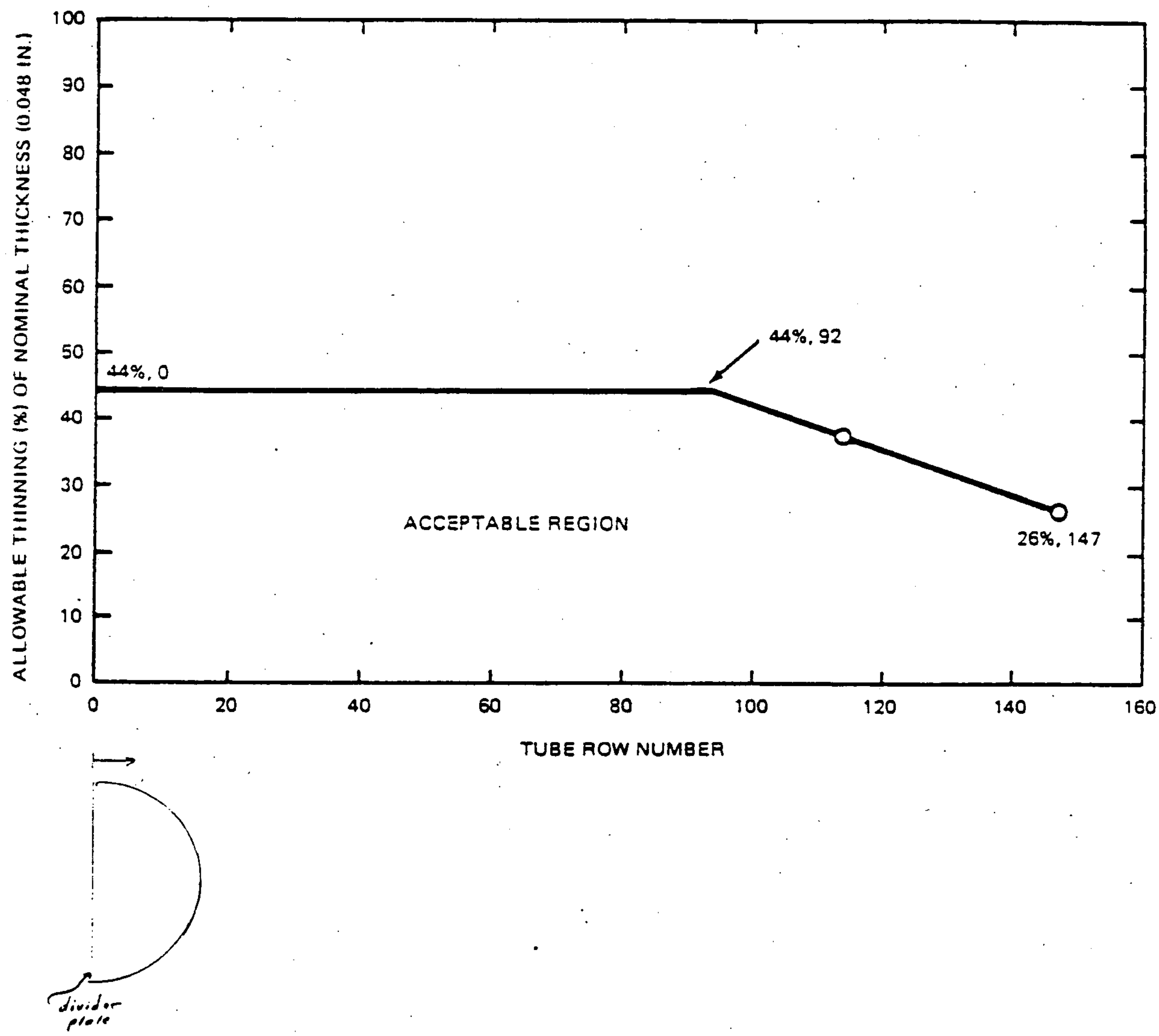

Figure 4.4-1
TUBE WALL THINNING ACCEPTANCE CRITERIA

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of Figure 4.4-1. Figure 4.4-1 was developed as a result of analyses performed for Primary Loop Pipe Break (PLPB) plus Design Basis Earthquake (DBE) and Main Steam Line Break (MSLB) plus DBE. These analyses examined families of tube rows as related to the number of vertical support grids. As horizontal tube spans become longer the loads become greater in spite of the increased number of tube supports. These results are controlling items in allowable tube wall thinning for tube rows 92 to 147. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10 CFR 50.72(b)(2)(i) prior to the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

DESCRIPTION OF PROPOSED CHANGES NPF-10-99 AND NPF-15-99
AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION.

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 is requested to clarify the Technical Specification requirements relating radiation monitors which support the containment purge isolation ESFAS function to improve consistency with the FSAR and Standard Technical Specifications (STS).

In conjunction with other proposed changes, this change also reflects the addition of the dedicated purge effluent monitors required by San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 License Conditions 2C(17) and 2C(15), respectively.

From the radiation monitoring standpoint, Standard Technical Specifications are functionally organized with separate specifications, operability requirements and actions for each monitoring function. The functional organization of the STS assumes that there are individual monitors to serve each function. When a given monitor serves more than one of the STS functions, as is the case at SONGS Units 2 and 3, the cross referencing of individual specification requirements tends to confuse and often makes the individual functionally related requirements overly restrictive. A review of the FSAR, Responses to TMI Action Plan, the Safety Evaluation Report, and related correspondence was conducted to determine which monitors have been credited to serve specific STS functions. As result of this review the following revisions are proposed:

1. Table 3.3-3, Item 12.b, Containment Airborne Monitors is revised to reflect FSAR credited functions. The primary function of the Containment Airborne Radiation Monitors (RT-7804-1 and RT-7807-2) is to actuate containment purge isolation in the event of a fuel handling accident in MODE 6 (FSAR Sections 7.3.1.1.5 and 11.5.2.1.4.5). In addition, the gaseous and particulate channels are credited with serving a reactor coolant system leak detection function required in MODES 1-4 by Specification 3/4.4.5 (FSAR Section 11.5.2.1.4.5). The proposed change

revises the applicability and ACTIONs to be consistent with the STS and appropriate for these functions. Prior to first refueling, containment airborne monitor, RT-7804-1, additionally serves the containment purge effluent monitoring function required by Specification 3.3.3.9. This function is reflected in Specification 3.3.3.9, Radioactive Gaseous Effluent Monitoring Instrumentation.

2. Table 3.3-3, Item 12.c, Containment Area Radiation, is revised to reflect FSAR credited function. The primary function of the containment area radiation monitors, RT-7856-1 and RT-7857-2, are to initiate containment purge isolation in the event of a fuel handling accident in MODE 6 or a small break LOCA in MODES 1-4 (FSAR Sections 7.3.1.1.5 and 12.3.4.3.1). These monitors are also credited with satisfying the NUREG-0737 Item II.E.4.2 requirement to isolate containment purge valves on a containment high radiation signal. The proposed change revises the applicability and ACTIONs and surveillance requirements to be consistent with Standard Technical Specifications, and to reflect these functions.

3. Table 3.3-4, Engineered Safety Feature Actuation System Instrumentation Trip Setpoints, Item 12.b, Airborne Radiation, and Item 12.c, Containment Area Radiation are revised to reflect the containment airborne and area monitors' ESFAS functions. As noted above, Containment Airborne Monitor RT 7804-1 currently satisfies the purge effluent monitoring requirements of Specification 3.3.3.9. Accordingly, the setpoints for this monitor are currently specified by the Offsite Dose Calculation Manual (ODCM). Prior to startup following the first refueling, Units 2 and 3 License Conditions 2.C(17) and 2.C(15), respectively, each require installation of a dedicated purge effluent monitor for their respective unit. On completion of these design changes, the containment airborne monitors will no longer serve the purge effluent monitoring function and it will no longer be appropriate to specify their setpoints in accordance with the ODCM. The proposed change requires that the setpoints be sufficiently high to prevent spurious alarms/trips but low enough to assure alarm/trip on an inadvertant release. This is consistent with the intent of the STS requirements for establishing setpoints.

   Two setpoints are specified for Item 12.c, the Containment Area Monitors. These setpoints correspond to the two functions noted above. The MODE 1-4 setpoint is consolidated from Specification 3.3.3.1, Radiation Alarm Monitoring Instrumentation. The 340 mR/hr allowable value results from the addition of a 5% of the setpoint allowance to account for needle width of this instrument's analog indicator. This is consistent with the practice used to establish the allowable values from trip setpoints of other radiation monitors with analog indicators in Table 3.3-4.

SAFETY ANALYSIS

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

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

   Response: No

   The function of containment purge isolation is to mitigate the radiological consequences of an in-containment fuel handling accident in MODE 6 and small break LOCA in MODES 1-4. The proposed change leaves this function intact. With the exception of the fuel handling accident noted above, no other previously analyzed accidents including the small break LOCA take credit for CPIS to mitigate the offsite dose consequences. However, as noted above, the radiation monitors which support CPIS are also credited in the FSAR with the performance of other non ESFAS functions. The proposed change is consistent with these other functions. However, from the standpoint of effects on the probability or consequences of previously evaluated accidents, no credit is taken for these other functions to mitigate the consequences of any FSAR Chapter 15 Accident Analysis other than the fuel handling accident as noted above. Therefore, the proposed change does not increase the probability or consequences of any previously evaluated accident.

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

   Response: No

   The proposed change clarifies the technical specification requirements for radiation monitoring instrumentation associated with containment purge isolation to be consistent with the FSAR. The proposed change does not affect the configuration or operation of the plant. It therefore 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 involves only a clarification of the technical specification requirements for radiation monitoring instrumentation associated with containment purge isolation. As noted above, the proposed change maintains requirements for this function consistent with the FSAR. Because the requirements remain the same as analyzed, no margin of safety is reduced by the proposed change.

The proposed clarification of containment purge isolation radiation monitoring requirements is similar to example (i) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it is essentially administrative in nature.

SAFETY AND SIGNIFICANT HAZARDS DETERMINATION

Based on the 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.

PSmith:0063F

ATTACHMENT "A"

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. FUEL HANDLING ISOLATION (FHIS)					
a. Manual (Trip Buttons)	2	1	1	**	16*#
b. Airborne Radiation					
i. Gaseous	2	1	1	**	16*#
ii. Particulate/Iodine	2	1	1	**	16*#
c. Automatic Actuation Logic	1/train	1	1	**	16*#
12. CONTAINMENT PURGE ISOLATION (CPIS)					
a. Manual (Trip Buttons)	2	1	1	6	17*#
b. Airborne Radiation					
i. Gaseous	2	1	1	All	17, 17a, 17b
ii. Particulate	2	1	1	All	17, 17a, 17b
iii. Iodine	2	1	1	All	17, 17b
c. Containment Area Radiation (Gamma)	2	1	1	6	17*#
d. Automatic Actuation Logic	1/train	1	1	All	17, 17a, 17b*#

Table 3.3-3 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the emergency (except as required by ACTIONS 14, 15) mode of operation.

ACTION 14 - With the number of channels OPERABLE one less than the total number of channels, restore the inoperable channel to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.

ACTION 15 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.

ACTION 16 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 17 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9. (Mode 6 only)

ACTION 17a - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1. (Mode 1, 2, 3, 4 only)

ACTION 17b - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.3.3.9. (At all times)

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
11. FUEL HANDLING ISOLATION (FHIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	$\leq 1.3 \times 10^2$ cpm**	$\leq 1.4 \times 10^2$ cpm**
ii. Particulate/Iodine	$\leq 5.7 \times 10^4$ cpm**	$\leq 6.0 \times 10^4$ cpm**
c. Automatic Actuation Logic	Not Applicable	Not Applicable
12. CONTAINMENT PURGE ISOLATION (CPIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	$\leq$ per ODCM	$\leq$ per ODCM
ii. Particulate	$\leq$ per ODCM	$\leq$ per ODCM
iii. Iodine	$\leq$ per ODCM	$\leq$ per ODCM
c. Containment Area Radiation (Gamma)	$\leq$ 2.4 mR/hr	$\leq$ 2.5 mR/hr
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

TABLE NOTATION

(1) Value may be decreased manually, to a minimum of greater than or equal to 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer and this value is maintained at less than or equal to 400 psia;* the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer is greater than or equal to 400 psia.

(2) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi;* the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

(4) Inverse time relay set value 3165V, trip will occur within the tolerances specified in Figure 3.3-1 for the range of bus voltages.

(5) Actuated equipment only; does not result in CIAS.

*Variable setpoints are for use only during <u>normal</u>, <u>controlled</u> plant heatups and cooldowns.

**Above normal background.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. FUEL HANDLING ISOLATION (FHIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	*
ii. Particulate/Iodine	S	R	M	*
c. Automatic Actuation Logic	N.A.	N.A.	R(3)	*
12. CONTAINMENT PURGE ISOLATION (CPIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	(2)	(2)	(2)	All
ii. Particulate	(2)	(2)	(2)	All
iii. Iodine	(2)	(2)	(2)	All
c. Containment Area Radiation (Gamma)	S	R	M	6
d. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) In accordance with Table 4.3-9 surveillance requirements for these instrument channels.

(3) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(4) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay.

(5) Actuated equipment only; does not result in CIAS.

* With irradiated fuel in the storage pool.

ATTACHMENT "B"

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. FUEL HANDLING ISOLATION (FHIS)					
a. Manual (Trip Buttons)	2	1	1	**	16*#
b. Airborne Radiation					
i. Gaseous	2	1	1	**	16*#
ii. Particulate/Iodine	2	1	1	**	16*#
c. Automatic Actuation Logic	1/train	1	1	**	16*#
12. CONTAINMENT PURGE ISOLATION (CPIS)					
a. Manual (Trip Buttons)	2	1	1	6	17b*#
b. Airborne Radiation (2RT7804-1 or 2RT7807- 2)					
i. Gaseous	2	1	1	1,2,3,4 6	17a 17b*#
ii. Particulate	2	1	1	1,2,3,4 6	17a 17b*#
iii. Iodine	2	1	1	6	17b*#
c. Containment Area Radiation(Gamma) (2RT7856-1 or 2RT7857- 2)	2	1	1	1,2,3,4 6	17 17b*#
d. Automatic Actuation Logic	1/train	1	1	1,2,3,4 6	17 17b*#

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Table 3.3-3 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the emergency (except as required by ACTIONS 14, 15) mode of operation.

ACTION 14 - With the number of channels OPERABLE one less than the total number of channels, restore the inoperable channel to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.

ACTION 15 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.

ACTION 16 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 17 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, operation may continue provided that the purge valves are maintained closed.

ACTION 17a - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1. (Mode 1, 2, 3, 4 only)

ACTION 17b - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
11. FUEL HANDLING ISOLATION (FHIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	$\leq 1.3 \times 10^2$ cpm**	$\leq 1.4 \times 10^2$ cpm**
ii. Particulate/Iodine	$\leq 5.7 \times 10^4$ cpm**	$\leq 6.0 \times 10^4$ cpm**
c. Automatic Actuation Logic	Not Applicable	Not Applicable
12. CONTAINMENT PURGE ISOLATION (CPIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	(6)(7)	(6)(7)
ii. Particulate	(6)(7)	(6)(7)
iii. Iodine	(6)(7)	(6)(7)
c. Containment Area Radiation (Gamma)	$\leq$ 325 mR/hr(MODES 1-4) $\leq$ 2.4mR/hr(MODE 6)	$\leq$ 340 mR/hr(MODES 1-4) $\leq$ 2.5 mR/hr(MODE 6)
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

TABLE NOTATION

(1) Value may be decreased manually, to a minimum of greater than or equal to 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer and this value is maintained at less than or equal to 400 psia;* the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer is greater than or equal to 400 psia.

(2) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi;* the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

(4) Inverse time relay set value 3165V, trip will occur within the tolerances specified in Figure 3.3-1 for the range of bus voltages.

(5) Actuated equipment only; does not result in CIAS.

(6) The trip setpoint shall be set sufficiently high to prevent spurious alarms/trips yet sufficiently low to assure an alarm/trip should an inadvertant release occur.

(7) Prior to the completion of DCP 53N, the setpoints for Containment Airborne Radiation Monitor 2RT-7804-1 shall be determined by the ODCM.

*Variable setpoints are for use only during normal, controlled plant heatups and cooldowns.

**Above normal background.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. FUEL HANDLING ISOLATION (FHIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	*
ii. Particulate/Iodine	S	R	M	*
c. Automatic Actuation Logic	N.A.	N.A.	R(3)	*
12. CONTAINMENT PURGE ISOLATION (CPIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	1,2,3,4,6
ii. Particulate	W	R	M	1,2,3,4,6
iii. Iodine	W	R	M	6
c. Containment Area Radiation (Gamma)	S	R	M	1,2,3,4,6
d. Automatic Actuation Logic	N.A.	N.A.	R (3)	1,2,3,4,6

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Deleted.

(3) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(4) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay.

(5) Actuated equipment only; does not result in CIAS.

* With irradiated fuel in the storage pool.

ATTACHMENT "C"

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. FUEL HANDLING ISOLATION (FHIS)					
a. Manual (Trip Buttons)	2	1	1	**	16*#
b. Airborne Radiation					
i. Gaseous	2	1	1	**	16*#
ii. Particulate/Iodine	2	1	1	**	16*#
c. Automatic Actuation Logic	1/train	1	1	**	16*#
12. CONTAINMENT PURGE ISOLATION (CPIS)					
a. Manual (Trip Buttons)	2	1	1	6	17*#
b. Airborne Radiation					
i. Gaseous	2	1	1	All	17, 17a, 17b
ii. Particulate	2	1	1	All	17, 17a, 17b
iii. Iodine	2	1	1	All	17, 17b
c. Containment Area Radiation (Gamma)	2	1	1	6	17*#
d. Automatic Actuation Logic	1/train	1	1	All	17, 17a, 17b*#

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Table 3.3-3 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the emergency (except as required by ACTIONS 14, 15) mode of operation.

ACTION 14 - With the number of channels OPERABLE one less than the total number of channels, restore the inoperable channel to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.

ACTION 15 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.

ACTION 16 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 17 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9. (MODE 6 only)

ACTION 17a - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1. (MODE 1, 2, 3, 4 only)

ACTION 17b - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.3.3.9. (At all times)

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
11. FUEL HANDLING ISOLATION (FHIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	$\leq 1.3 \times 10^2$ cpm**	$\leq 1.4 \times 10^2$ cpm**
ii. Particulate/Iodine	$\leq 5.7 \times 10^4$ cpm**	$\leq 6.0 \times 10^4$ cpm**
c. Automatic Actuation Logic	Not Applicable	Not Applicable
12. CONTAINMENT PURGE ISOLATION (CPIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	$\leq$ per ODCM	$\leq$ per ODCM
ii. Particulate	$\leq$ per ODCM	$\leq$ per ODCM
iii. Iodine	$\leq$ per ODCM	$\leq$ per ODCM
c. Containment Area Radiation (Gamma)	$\leq$ 2.4 mR/hr	$\leq$ 2.5 mR/hr
d. Automatic Actuation Logic	Not Applicable	Not Applicable

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TABLE 3.3-4 (Continued)

TABLE NOTATION

(1) Value may be decreased manually, to a minimum of greater than or equal to 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer and this value is maintained at less than or equal to 400 psia;* the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer is greater than or equal to 400 psia.

(2) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi;* the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

(4) Inverse time relay set value 3165V, trip will occur within the tolerances specified in Figure 3.3-1 for the range of bus voltages.

(5) Actuated equipment only; does not result in CIAS.

*Variable setpoints are for use only during <u>normal</u>, <u>controlled</u> plant heatups and cooldowns.

**Above normal background.

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. FUEL HANDLING ISOLATION (FHIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	*
ii. Particulate/Iodine	S	R	M	*
c. Automatic Actuation Logic	N.A.	N.A.	R(3)	*
12. CONTAINMENT PURGE ISOLATION (CPIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	(2)	(2)	(2)	All
ii. Particulate	(2)	(2)	(2)	All
iii. Iodine	(2)	(2)	(2)	All
c. Containment Area Radiation (Gamma)	S	R	M	6
d. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) In accordance with Table 4.3-9 Surveillance Requirements for these instrument channels.

(3) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(4) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay.

(5) Actuated equipment only; does not result in CIAS.

* With irradiated fuel in the storage pool.

ATTACHMENT "D"

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. FUEL HANDLING ISOLATION (FHIS)					
a. Manual (Trip Buttons)	2	1	1	**	16*#
b. Airborne Radiation					
i. Gaseous	2	1	1	**	16*#
ii. Particulate/Iodine	2	1	1	**	16*#
c. Automatic Actuation Logic	1/train	1	1	**	16*#
12. CONTAINMENT PURGE ISOLATION (CPIS)					
a. Manual (Trip Buttons)	2	1	1	6	17b*#
b. Airborne Radiation (3RT-7804-1 or 3RT-7807-2)					
i. Gaseous	2	1	1	1,2,3,4 6	17a 17b*#
ii. Particulate	2	1	1	1,2,3,4 6	17a 17b*#
iii. Iodine	2	1	1	6	17b*#
c. Containment Area Radiation (Gamma) (3RT-7856-1 or 3RT-7857-2)	2	1	1	1,2,3,4 6	17 17b*#
d. Automatic Actuation Logic	1/train	1	1	1,2,3,4 6	17 17b*#

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Table 3.3-3 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the emergency (except as required by ACTIONS 14, 15) mode of operation.

ACTION 14 - With the number of channels OPERABLE one less than the total number of channels, restore the inoperable channel to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.

ACTION 15 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.

ACTION 16 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 17 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, operation may continue provided that the purge valves are maintained closed.

ACTION 17a - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1. (MODE 1, 2, 3, 4 only)

ACTION 17b - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere.

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
11. FUEL HANDLING ISOLATION (FHIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	$\leq 1.3 \times 10^2$ cpm**	$\leq 1.4 \times 10^2$ cpm**
ii. Particulate/Iodine	$\leq 5.7 \times 10^4$ cpm**	$\leq 6.0 \times 10^4$ cpm**
c. Automatic Actuation Logic	Not Applicable	Not Applicable
12. CONTAINMENT PURGE ISOLATION (CPIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	(6)(7)	(6)(7)
ii. Particulate	(6)(7)	(6)(7)
iii. Iodine	(6)(7)	(6)(7)
c. Containment Area Radiation (Gamma)	$\leq$ 325 mR/hr(MODES 1-4) $\leq$ 2.4 mR/hr(MODE 6)	$\leq$ 340 mR/hr (MODES 1-4) $\leq$ 2.5 mR/hr (MODE 6)
d. Automatic Actuation Logic	Not Applicable	Not Applicable

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TABLE 3.3-4 (Continued)

TABLE NOTATION

(1) Value may be decreased manually, to a minimum of greater than or equal to 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer and this value is maintained at less than or equal to 400 psia;* the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer is greater than or equal to 400 psia.

(2) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi;* the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

(4) Inverse time relay set value 3165V, trip will occur within the tolerances specified in Figure 3.3-1 for the range of bus voltages.

(5) Actuated equipment only; does not result in CIAS.

(6) The trip setpoint shall be set sufficiently high to prevent spurious alarms/trips yet sufficiently low to assure an alarm/trip should an inadvertant release occur.

(7) Prior to the completion of DCP 53N, the setpoints for Containment Airborne Radiation Monitor 3RT-7804-1 shall be determined by the ODCM.

*Variable setpoints are for use only during normal, controlled plant heatups and cooldowns.

**Above normal background.

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. FUEL HANDLING ISOLATION (FHIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	*
ii. Particulate/Iodine	S	R	M	*
c. Automatic Actuation Logic	N.A.	N.A.	R(3)	*
12. CONTAINMENT PURGE ISOLATION (CPIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	1,2,3,4,6
ii. Particulate	W	R	M	1,2,3,4,6
iii. Iodine	W	R	M	6
c. Containment Area Radiation (Gamma)	S	R	M	1,2,3,4,6
d. Automatic Actuation Logic	N.A.	N.A.	R (3)	1,2,3,4,6

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Deleted.

(3) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(4) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay.

(5) Actuated equipment only; does not result in CIAS.

* With irradiated fuel in the storage pool.

DESCRIPTION OF PROPOSED CHANGES NPF-10-100 AND NPF-15-100
AND SAFETY ANALYSIS.

This is a request to revise Technical Specification 3/4.3.3.1, RADIATION ALARM MONITORING INSTRUMENTATION.

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 is required to clarify the techinical specification requirements relating to radiation monitors which support alarm functions, and to improve consistency with the FSAR, Standard Technical Specifications (STS) and other technical specifications covering other functions served by the same instruments. Additionally, the proposed change adds flexibility to the ACTION statements and revises plant vent stack monitoring requirements and the applicability for the condenser evacuation system monitors. To this end the following revisions are made:

1. Consistent with the STS, the word "alarm" is deleted from the title of Specificaton 3/4.3.3.1 and elsewhere in the specification where it is used in this context and the words "alarm/trip" are substituted for the word "alarm" where it is used in the context of setpoint.

2. Specification 3/4.3.2, Engineered Safety Features Actuation System (ESFAS), delineates functional requirements for radiation monitors which support the Control Room Isolation Signal (CRIS), the Fuel Handling Isolation Signal (FHIS) and the Containment Purge Isolation Signal (CPIS) functions. Functional requirements for the Radiation monitors which support these ESFAS functions are also specified by Specifiation 3/4.3.3.1. The proposed changes to Items 1.b, 2.a, 2.b and 2.c of Tables 3.3-6 and 4.3-3 make Specification 3/4.3.3.1 consistent with Specification 3/4.3.2 by directly referencing the 3/4.3.2 setpoints and ACTION requirements. The changes to Items 1.b and 2.b are also consistent with Proposed Changes NPF-10-99 and NPF-15-99 which revise the Specification 3/4.3.2 requirements relating to CPIS to be consistent with FSAR commitments.

3. To satisfy NUREG 0737 requirements, Containment High Range Area Monitors, Main Steam Line Area Monitors, Plant Vent/Purge Stack and Condenser Evacuation System wide range noble gas monitors were installed. Currently, the operability requirements for these monitors are

distributed between Specification 3/4.3.3.6, Accident Monitoring Instrumentation and Specification 3/4.3.3.1, Radiation Monitoring Alarm Instrumentation. The proposed change consolidates the requirements for these 0737 radiation monitors in Specification 3/4.3.3.1. Another proposed change (NPF-10-101/NPF-15-101) deletes the requirements for these monitors from Specification 3/4.3.3.6. The consolidation of the requirements for the NUREG 0737 monitors in Specification 3/4.3.3.1 will reduce the complexity of the specifications. Consolidation of the requirements for NUREG 0737 radiation monitors is consistant with the STS and St. Lucie Unit 2 Technical Specifications.

Specifiation 3.3.3.6 currently requires both wide range plant vent stack monitors (2 RT-7865-1 and 3 RT-7865-1) to be operable in MODES 1-3. In addition to consolidating the NUREG 0737 radiation monitoring requirements in Specification 3.3.3.1, the proposed change reduces the required number of wide range plant vent stack monitors from two to one. This is acceptable because the Unit 2 and Unit 3 plant vent stacks are not totally independent effluent paths. Exhaust from the shared auxiliary buildings and the two fuel handling buildings are mixed in a common plenum and released via the Units 2 and 3 plant vent stacks. Plant vent stack monitor 2/3 RT-7808-1 provides noble gas monitoring capability for normal operation and anticipated operational occurrences. Wide range noble gas effluent monitors 2 RT--7865-1 and 3 RT-7865-1 for the Units 2 and 3 plant vent stacks, respectively provide post accident noble gas monitoring capability and can monitor effluents from either the vent stack or the purge stack of the associated unit. Although only approximately one-half of the plant vent stack effluent is monitored by each of the RT-7865 wide range noble gas monitors, sufficient data has been accumulated to provide a consistently conservative estimate of the releases from one plant vent stack based on the readings from the other unit's plant vent stack monitor. It is desireable to have both plant vent stack monitors (2 RT-7865-1 and 3 RT-7865-1) operable. However, because post accident plant vent stack releases can be tracked reliably by one of the plant vent stack wide range monitors, a minimum of one channel is required to be operable in MODES 1-3. In MODE 4, in lieu of th wide range plant vent stack monitors, the plant vent stack noble gas monitoring function can be satisfied by 2/3 RT-7808-1, the normal range instrument, because both the probability of occurrence of design bases accidents and consequent effluent activity are likely to be significantly lower in MODE 4.

Each of the wide range plant vent stack monitors (2 RT-7865-1, 3 RT-7865-1) also serves the post accident wide range purge effluent monitoring function for its associated unit. With the above minimum channels operable requirement, a unit can operate without wide range post accident purge effluent monitoring capability necessarily being available. Technical Specification requirements for containment and purge isolation system operability ensure that the containment purge stack would not be an uncontrolled release path in the event that a design basis accident occurs. Post accident purging would be a

controlled action. Implicit in the control of post accident purging is the requirement to have adequate monitoring capability prior to initiating a post accident containment purge.

4. The proposed change revises ACTION 18 to eliminate the current reference to ACTIONs 20 and 21 of Specification 3.3.3.6. In addition, ACTIONs 18 and 19 are revised to allow more time to restore inoperable channels to OPERABLE status. ACTION 18, i.e. ACTION 20 by cross reference, currently allows 7 days to restore an inoperable channel. ACTION 20 applies to many instruments which would be used to mitigate the consequences of a design basis accident, in addition to the radiation monitoring channels currently covered by ACTIONs 18 and 20. It should be noted that the radiation monitoring channels do not directly contribute to the mitigation of consequences of a design basis accident in the same sense as the other accident monitoring instrumentation listed in Table 3.3-10. Therefore, less severe ACTIONs are justifiable for radiation monitoring channels which are not directly used in mitigating of the consequences of design basis accidents.

ACTION 18 (and 20) which applies to the area monitors-listed in Table 3.3-6, currently allow 7 days to restore an inoperable instrument or shutdown. The high range area monitors have proven to be difficult to trouble-shoot and in the past it has taken very close to 7 days to repair an inoperable instrument. The difficulty associated with trouble-shooting of these instruments is directly related to the requirement for these instruments to be environmentally qualified to operate in the postulated high post accident radiation fields. This requirement precludes the use of pre-amplifiers located at the detectors. As result only the very small currents (on the order of a few pico amps) generated by the detectors are carried by the cables to the instrument electronics located in low radiation areas. Because of the small currents involved, trouble-shooting is difficult and time consuming. The proposed change to ACTION 18 allows 30 days to restore an inoperable instrument to operable status. Consistent with past experience this change would significantly reduce the possibility of a shutdown.

ACTION 19 is clarified with respect to the special reporting requirements. ACTION 19 part 1) currently states that a pre-planned alternate method of monitoring be initiated if the channel is not returned to operable status within 72 hours. If the instrument is returned to operable status within 72 hours no action is required. However, part 2) of ACTION 19 requires a Special Report to be submitted within 14 days following the event. The word "event" is ambiguous in that the event could be either the inoperability of the channel or the initiation of the pre-planned alternate. If "event" refers to the inoperability, then in a situation where the channel was restored to operable status within 72 hours and no pre-planned alternate was initiated, the special report outlining the action taken, and plans and schedule for restoring operability is meaningless. The proposed change clarifies ACTION 19 to require a special report only if the inoperability is not corrected within 72 hours and the pre-planned alternate is initiated.

5. The proposed change revises the Table 3.3-6 applicability for the plant vent stack and condenser evacuation system noble gas monitors. The current all modes applicability reflects effluent monitoring requirements. Effluent monitoring instrumentation requirements are more appropriately specified in Specification 3.3.3.9. The proposed change reduces the applicability for the plant vent stack and condenser evacuation system monitors from the accident monitoring instrumentation standpoint. This is consistent with the Standard Technical Specifications for radiation monitoring instrumentation. The effect of this will be to relieve more stringent accident monitoring requirements from being applied in MODES where only effluent monitoring is the primary concern.

   The condenser evacuation system is monitored because it is a potential gaseous radioactive effluent release path during normal plant operation due to primary to secondary leakage within the allowable limits and in the event of a steam generator tube rupture. However, when the Main Steam Isolation Valves (MSIV's) and main steam isolating valve bypass valves are fully closed, the condenser is isolated from its potential source of gaseous activity and, therefore, is not a potential gaseous radioactive effluent release path when these conditions are met. Accordingly, the proposed change requires noble gas monitoring for the condenser evacuation system in MODES 1-4 only when the MSIV's and MSIV bypass valves are open. A corresponding proposed change (NPF-10-102 and NPF-15-102) makes a similar ajustment to the condenser evacuation system monitoring applicability of Specification 3.3.3.9.

6. The proposed change identifies the required radiation monitoring instrumentation by instrument number to improve clarity of the Specification.

Safety analysis

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

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

   Response: No.

   The proposed change affects the technical specification requirements for certain radiation monitoring instrumentation. With the exception of the containment purge isolation area monitors (Table 3.3-6 Item 1.b) and the containment airborne monitors (Table 3.3-6 Item 2.b), the proposed change does not affect the requirements for any radiation monitors credited in the mitigation of the consequences of any previously evaluated accident. The containment purge isolation area monitors and the containment airborne monitors support the ESFAS, containment purge isolation function.

This function is intended to mitigate the radiological consequences of an in-containment fuel handling accident in MODE 6 and small break LOCA in MODES 1-4. The proposed change leaves this function intact. With the exception of the fuel handling accident noted above, no other previously analyzed accidents including the small break LOCA take credit for CPIS to mitigate the offsite dose consequences.

Therefore, the proposed change does not increase the probability or consequences of any previously evaluated accident.

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

   Response: No.

   The proposed change clarifies the technical specification requirements for radiation monitoring instrumentation to be consistent with the FSAR. The proposed change does not affect the configuration or operation of the plant. It therefore 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 clarifies the technical specifiation requirements for radiation monitoring instrumentation. The proposed change reduces operability requirements for noble gas radiation monitors on the plant vent stacks and condenser evacuation system. It also increases the time allowed in ACTION statements to accommodate repair, maintenance and calibration of the affected instruments. Although the proposed change involves a reduction in requirements, it maintains the ability to provide the required post-accident assessment of radioactive gaseous releases and radiation conditions within the plant. The radiation monitors affected by the proposed change are not credited with the mitigation of any previously evaluated accident, with the exception of those supporting CPIS. The requirements for the CPIS related monitors are maintained by the proposed change to be consistent with the FSAR. The proposed change does not affect the consequences of any previously evaluated accident. Therefore, no margin of safety is reduced.

48 FR 14864 dated April 6, 1983 provided examples of amendments that are not likely to involve a significant hazards consideration. In comparison with these examples, Items 1 and 6 of the description section are similar to example (i) in that they are editorial in nature. Items 2, 3, 4 and 5 involve some reduction in existing technical specification requirements. Although these items do not increase the probablity or consequences of any previously analyzed accident, they would likely be considered to be most similar to example (vi) in that the reduction in technical specification requirements may be perceived to insignificantly reduce in some way a safety margin.

Safety And Significant Hazards Determination

Based on the above 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.

PSmith:0913

ATTACHMENT "A"

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING ALARM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring alarm instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.*

ACTION:

a. With a radiation monitoring channel alarm setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring alarm channels inoperable, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring alarm instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

*See Special Test Exception 3.10.5.

TABLE 3.3-6

RADIATION MONITORING ALARM INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
1. Area Monitors					
a. Containment - High Range	2	1, 2, 3	10 R/hr	$1\text{-}10^8$ R/hr	18
		4	10 R/hr		19
b. Containment - Purge Isolation	1	1, 2, 3, 4	$\leq$ 325 mR/hr	$10^{-1}\text{-}10^5$ mR/hr	19
		6	#		(a)
c. Main Steam Line	1/line	1, 2, 3	1 mR/hr (low); 1 R/hr (high)	$10^{-1}\text{-}10^4$ mR/hr;	18
		4	1 mR/hr (low); 1 R/hr (high)		19
2. Process Monitors					
a. Fuel Storage Pool Airborne					
i. Gaseous	1	*	#	10^1 -10^7 cpm	(d)
ii. Particulate/Iodine	1	*	#	10^1 -10^7 cpm	(d)
b. Containment Airborne					
i. Gaseous	1	All	Per ODCM	10^1 -10^7 cpm	(a)(b)(c)
ii. Particulate	1	All	Per ODCM	10^1 -10^7 cpm	(a)(b)(c)
iii. Iodine	1	All	Per ODCM	10^1 -10^7 cpm	(a)(c)
c. Control Room Airborne					
i. Particulate/Iodine	1	All	#	10^1 -10^7 cpm	(e)
ii. Gaseous	1	All	#	10^1 -10^7 cpm	(e)

TABLE 3.3-6 (Continued)

RADIATION MONITORING ALARM INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
3. Noble Gas Monitors					
a. Plant Vent Stack	1	All	Per ODCM	$10^1 - 10^7$ cpm	19, (c)
b. Condenser Evacuation System	1	All	Per ODCM	$10^1 - 10^7$ cpm	19, (c)

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 18 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.3.3.6.

ACTION 19 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

#In accordance with Engineered Safety Feature trip value specified by Table 3.3-4.

*With irradiated fuel in the storage pool.

(a) In accordance with Table 3.3-3 - ACTION 17.

(b) In accordance with Table 3.3-3 - ACTION 17a.

(c) In accordance with Table 3.3-3 - ACTION 17b.

(d) In accordance with Table 3.3-3 - ACTION 16.

(e) In accordance with Table 3.3-3 - ACTION 13.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Area Monitors				
a. Containment - High Range	S	R	M	1, 2, 3, 4
b. Containment - Purge Isolation	S #	R #	M #	1, 2, 3, 4 6
c. Main Steam Line	S	R	M	1, 2, 3, 4
2. Process Monitors				
a. Fuel Storage Pool Airborne				
i. Gaseous	#	#	#	*
ii. Particulate/Iodine	#	#	#	*
b. Containment Airborne				
i. Gaseous	@	@	@	All
ii. Particulate	@	@	@	All
iii. Iodine	@	@	@	All
c. Control Room Airborne				
i. Particulate	#	#	#	All
ii. Gaseous	#	#	#	All

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
PROCESS MONITORS (Continued)				
3. Noble Gas Monitors				
a. Plant Vent Stack	@	@	@	All
b. Condenser Evacuation System	@	@	@	All

NOTES:

#In accordance with Table 4.3-2 surveillance requirements for these instrument channels.

*With irradiated fuel in the storage pool.

@In accordance with Table 4.3-9 surveillance requirements for these instrument channels.

ATTACHMENT "B"

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.*

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

*See Special Test Exception 3.10.5.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM /TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Area Monitors					
a. Containment - High Range (2RT-7820-1 and 2RT-7820-2)	2	1, 2, 3	10 R/hr	$1-10^{8}$ R/hr	18, 18a
		4	10 R/hr		19
b. Containment - Purge Isolation (2RT-7856-1 or 2RT-7857-2)	1	1, 2, 3, 4	#	$10^{-1}-10^{5}$ mR/hr	17
		6	#		17b
c. Main Steam Line A Channel consists of 2RT-7874A and 2RT-7875A or 2RT-7874B and 2RT-7875B	1/line	1, 2, 3	1 mR/hr (low); 1 R/hr (high)	$10^{-1}-10^{4}$ mR/hr;	18
		4	1 mR/hr (low); 1 R/hr (high)		19
2. Process Monitors					
a. Fuel Storage Pool Airborne (2 RT-7822-1 or 2 RT-7823-2)					
i. Gaseous	1	*	#	$10^{1}-10^{7}$ cpm	16
ii. Particulate/Iodine	1	*	#	$10^{1}-10^{7}$ cpm	16
b. Containment Airborne (2RT-7804-1 or 2RT-7807-2)					
i. Gaseous	1	1,2,3,4	#	$10^{1}-10^{7}$ cpm	17a
		6	#		17b
ii. Particulate	1	1,2,3,4	#	$10^{1}-10^{7}$ cpm	17a
		6	#		17b
iii. Iodine	1	6	#	$10^{1}-10^{7}$ cpm	17b
c. Control Room Airborne (2/3 RT-7824-1 or 2/3 RT-7825-2)					
i. Particulate	1	All	#	$10^{1}-10^{7}$ cpm	13
ii. Gaseous	1	All	#	$10^{1}-10^{7}$ cpm	13

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
3. Noble Gas Monitors					
a. Plant Vent Stack					
Wide Range(2RT-7865-1 or 3RT-7865-1)	1	1,2,3	Per ODCM	10^{-7}-10^{5} uCi/cm^{3}	19
Normal Range(2RT-7865-1, 3RT-7865-1 or 2/3RT-7808)	1	4	Per ODCM	10^{-6}-10^{-1}uCi/cm^{3}	19
b. Condenser Evacuation System					
Wide Range(2RT-7870-1)	1	1,2,3(1)	Per ODCM	10^{-7}-10^{5} uCi/cm^{3}	19
Normal Range(2RT-7818 or 2RT-7870-1)	1	4 (1)	Per ODCM	10^{-6}-10^{2} uCi/cm^{3}	19

(1) With any main steam isolation valve and/or any main steam isolating valve bypass valve not fully closed.

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 13 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the emergency (except as required by ACTIONS 14, 15) mode of operation.

ACTION 16 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 17 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, operation may continue provided that the purge valves are maintained closed.

ACTION 17a - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1. (Mode 1, 2, 3, 4 only)

ACTION 17b - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere.

ACTION 18 - With the number of channels OPERABLE one less than Minimum Channels OPERABLE requirement, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 18a - With both channels inoperable, restore the inoperable channel(s) to OPERABLE status within 48 hours, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 19 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 Hours,or:

1) Initiate the preplanned and alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following initiation of the pre-planned alternate outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

#In accordance with Engineered Safety Feature trip value specified by Table 3.3-4
* With irradiated fuel in the storage pool.

ACTIONS 13, 16, 17, 17a and 17b are repeated from Table 3.3-3 for reference.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Area Monitors				
a. Containment - High Range (2RT-7820-1, 2RT-7820-2)	S	R	M	1, 2, 3, 4
b. Containment - Purge Isolation (2RT-7856-1, 2RT-7857-2)	S	R	M	1, 2, 3, 4, 6
c. Main Steam Line (2RT-7874A, 2RT-7875A, 2RT-7874B, 2RT-7875B)	S	R	M	1, 2, 3, 4
2. Process Monitors				
a. Fuel Storage Pool Airborne (2RT-7822-1, 2RT-7823-2)				
i. Gaseous	#	#	#	*
ii. Particulate/Iodine	#	#	#	*
b. Containment Airborne (2RT-7804-1, 2RT-7807-2)				
i. Gaseous	#	#	#	1,2,3,4,6
ii. Particulate	#	#	#	1,2,3,4,6
iii. Iodine	#	#	#	6
c. Control Room Airborne (2/3RT-7824-1, 2/3RT-7825-2)				
i. Particulate	#	#	#	All
ii. Gaseous	#	#	#	All

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
PROCESS MONITORS (Continued)				
3. Noble Gas Monitors				
a. Plant Vent Stack (2RT-7865-1, 3RT-7865-1, 2/3RT-7808)	D	R	Q	1,2,3,4
b. Condenser Evacuation System (2RT-7870-1, 2RT-7818-1)	D	R	Q	1,2,3,4(1)

NOTES:

#In accordance with Table 4.3-2 surveillance requirements for these instrument channels.

*With irradiated fuel in the storage pool.

(1) With any main steam isolation valve and/or any main steam isolating valve bypass valve not fully closed.

ATTACHMENT "C"

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING ALARM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring alarm instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.*

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring alarm channels inoperable, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring alarm instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

*Continuous monitoring and sampling of the containment purge exhaust directly from the purge stack shall be provided for the low and high volume (8-inch and 42-inch) containment purge prior to startup following the first refueling outage. Containment airborne monitor 3RT-7804-1 or 3RT-7807-2 and associated sampling media shall perform these functions prior to initial criticality. From initial criticality to the startup following the first refueling outage containment airborne monitor 3RT-7804-1 and associated sampling media shall perform the above required functions.

TABLE 3.3-6

RADIATION MONITORING ALARM INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
1. Area Monitors					
a. Containment - High Range	2	1, 2, 3	10 R/hr	$1-10^{8}$ R/hr	18
		4	10 R/hr		19
b. Containment - Purge Isolation	1	1, 2, 3, 4	≤ 325 mR/hr	$10^{-1}-10^{5}$ mR/hr	19
		6	#		(a)
c. Main Steam Line	1/line	1, 2, 3	1 mR/hr (low); 1 R/hr (high)	$10^{-1}-10^{4}$ mR/hr;	18
		4	1 mR/hr (low); 1 R/hr (high)		19
2. Process Monitors					
a. Fuel Storage Pool Airborne					
i. Gaseous	1	*	#	$10^{1}-10^{7}$ cpm	(d)
ii. Particulate/Iodine	1	*	#	$10^{1}-10^{7}$ cpm	(d)
b. Containment Airborne					
i. Gaseous	1	All	Per ODCM	$10^{1}-10^{7}$ cpm	(a)(b)(c)
ii. Particulate	1	All	Per ODCM	$10^{1}-10^{7}$ cpm	(a)(b)(c)
iii. Iodine	1	All	Per ODCM	$10^{1}-10^{7}$ cpm	(a)(c)
c. Control Room Airborne					
i. Particulate/Iodine	1	All	#	$10^{1}-10^{7}$ cpm	(e)
ii. Gaseous	1	All	#	$10^{1}-10^{7}$ cpm	(e)

TABLE 3.3-6 (Continued)

RADIATION MONITORING ALARM INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
3. Noble Gas Monitors					
a. Plant Vent Stack	1	All	Per ODCM	$10^1 - 10^7$ cpm	19, (c)
b. Condenser Evacuation System	1	All	Per ODCM	$10^1 - 10^7$ cpm	19, (c)

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 18 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.3.3.6.

ACTION 19 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

#In accordance with Engineered Safety Feature trip value specified by Table 3.3-4.

*With irradiated fuel in the storage pool.

(a) In accordance with Table 3.3-3 - ACTION 17.

(b) In accordance with Table 3.3-3 - ACTION 17a.

(c) In accordance with Table 3.3-3 - ACTION 17b.

(d) In accordance with Table 3.3-3 - ACTION 16.

(e) In accordance with Table 3.3-3 - ACTION 13.

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TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Area Monitors				
a. Containment - High Range	S	R	M	1, 2, 3, 4
b. Containment - Purge Isolation	S	R	M	1, 2, 3, 4
	#	#	#	6
c. Main Steam Line	S	R	M	1, 2, 3, 4
2. Process Monitors				
a. Fuel Storage Pool Airborne				
i. Gaseous	#	#	#	*
ii. Particulate/Iodine	#	#	#	*
b. Containment Airborne				
i. Gaseous	@	@	@	All
ii. Particulate	@	@	@	All
iii. Iodine	@	@	@	All
c. Control Room Airborne				
i. Particulate	#	#	#	All
ii. Gaseous	#	#	#	All

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
PROCESS MONITORS (Continued)				
3. Noble Gas Monitors				
a. Plant Vent Stack	@	@	@	All
b. Condenser Evacuation System	@	@	@	All

NOTES:

#In accordance with Table 4.3-2 surveillance requirements for these instrument channels.

*With irradiated fuel in the storage pool.

@In accordance with Table 4.3-9 surveillance requirements for these instrument channels.

ATTACHMENT "D"

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.*

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm /trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

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

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

*Continuous monitoring and sampling of the containment purge exhaust directly from the purge stack shall be provided for the low and high volume (8-inch and 42-inch) containment purge prior to startup following the first refueling outage. Containment airborne monitor 3RT-7804-1 or 3RT-7807-2 and associated sampling media shall perform these functions prior to initial criticality. From initial criticality to the startup following the first refueling outage containment airborne monitor 3RT-7804-1 and associated sampling media shall perform the above required functions.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Area Monitors					
a. Containment - High Range (3RT-7820-1 and 3RT-7820-2	2	1, 2, 3 4	10 R/hr 10 R/hr	$1-10^8$ R/hr	18, 18a 19
b. Containment - Purge Isolation (3RT-7856-1 or 3RT-7857-2)	1	1, 2, 3, 4 6	# #	$10^{-1}-10^5$ mR/hr	17 17b
c. Main Steam Line A channel consists of 3RT-7874A and 3RT-7875A or 3RT-7874B and 3RT-7875B	1/line	1, 2, 3 4	1 mR/hr (low); 1 R/hr (high) 1 mR/hr (low); 1 R/hr (high)	$10^{-1}-10^4$ mR/hr;	18 19
2. Process Monitors					
a. Fuel Storage Pool Airborne (3RT-7822-1 or 3RT-7823-2)					
i. Gaseous	1	*	#	10^1-10^7 cpm	16
ii. Particulate/Iodine	1	*	#	10^1-10^7 cpm	16
b. Containment Airborne (3RT-7804-1 or 3RT-7807-2)					
i. Gaseous	1	1,2,3,4 6	# #	10^1-10^7 cpm	17a 17b
ii. Particulate	1	1,2,3,4 6	# #	10^1-10^7 cpm	17a 17b
iii. Iodine	1	6	#	10^1-10^7 cpm	17b
c. Control Room Airborne (2/3 RT-7824-1 or 2/3 RT-7825-2)					
i. Particulate/Iodine	1	All	#	10^1-10^7 cpm	13
ii. Gaseous	1	All	#	10^1-10^7 cpm	13

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
3. Noble Gas Monitors					
a. Plant Vent Stack					
Wide Range(2RI-7865-1 or 3RI-7865-1)	1	1,2,3	Per ODCM	10^{-7}-10^{5} uCi/cm3	19
Normal Range(2/3RI-7808 or 2RI-7865-1 or 3RI-7865-1)	1	4	Per ODCM	10^{-6}-10^{-1}uCi/cm^{3}	19
b. Condenser Evacuation System					
Wide Range(3RI-7870-1)	1	1,2,3(1)	Per ODCM	10^{-7}-10^{5}uCi/cm^{3}	19
Normal Range(3RI-7818 or 3RI-7870-1)	1	4 (1)	Per ODCM	10^{-6}-10^{2}uCi/cm^{3}	19

(1) With any main steam line isolation valve and/or any main steam isolating valve bypass valve not fully closed.

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TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 13 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the emergency (except as required by ACTIONS 14, 15) mode of operation.

ACTION 16 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 17 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, operation may continue provided that the purge valves are maintained closed.

ACTION 17a - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1. (Mode 1, 2, 3, 4 only)

ACTION 17b - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere.

ACTION 18 - With the number of channels OPERABLE one less than Minimum Channels OPERABLE requirement, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 18a - With both channels inoperable, restore the inoperable channel(s) to OPERABLE status within 48 hours, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 19 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 Hours, or:

1) Initiate the preplanned and alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following initiation of the pre-planned alternate outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

#In accordance with Engineered Safety Feature trip value specified by Table 3.3-4
* With irradiated fuel in the storage pool.
ACTIONS 13, 16, 17, 17a and 17b are repeated from Table 3.3-3 for reference.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Area Monitors				
a. Containment - High Range (3RT-7820-1, 3RT-7820-2)	S	R	M	1, 2, 3, 4
b. Containment - Purge Isolation (3RT-7856-1, 3RT-7857-2)	S	R	M	1, 2, 3, 4 ,6
c. Main Steam Line (3RT-7874A, 3RT-7875A, 3RT-7874B, 3RT-7875B)	S	R	M	1, 2, 3, 4
2. Process Monitors				
a. Fuel Storage Pool Airborne (3RT-7822-1, 3RT-7823-2)				
i. Gaseous	#	#	#	*
ii. Particulate/Iodine	#	#	#	*
b. Containment Airborne (3RT-7804-1, 3RT-7807-2)				
i. Gaseous	#	#	#	1,2,3,4,6
ii. Particulate	#	#	#	1,2,3,4,6
iii. Iodine	#	#	#	6
c. Control Room Airborne (2/3RT-7824-1, 2/3RT-7825-2)				
i. Particulate	#	#	#	All
ii. Gaseous	#	#	#	All

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
PROCESS MONITORS (Continued)				
3. Noble Gas Monitors				
a. Plant Vent Stack (2/3 RT-7808, 2RT-7865-1,3RT-7865-1)	D	R	Q	1,2,3,4
b. Condenser Evacuation System (3RT-7818, 3RT-7870-1)	D	R	Q	1,2,3,4 (1)

NOTES:

In accordance with Table 4.3-2 surveillance requirements for these instrument channels.

* With irradiated fuel in the storage pool.

(1) With any main steam isolation valve and/or any main steam isolating valve bypass valve not fully closed.

DESCRIPTION OF PROPOSED CHANGES NPF-10-101 AND NPF-15-101 AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3.3.3.6, ACCIDENT MONITORING INSTRUMENTATION.

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 is required to clarify the technical specification requirements for radiation monitoring instrumentation and to improve consistency with the Standard Technical Specifications (STS) and other technical specifications and proposed changes.

The proposed change deletes from Specification 3.3.3.6 those radiation monitors listed in Table 3.3-10 as items 19, 20, 21 and 22. These wide range radiation monitors were installed to satisfy NUREG-0737 requirements. Consistent with STS, the requirements for these wide range monitors are more appropriately delineated in Specification 3.3.3.1, RADIATION MONITORING INSTRUMENTATION. Another proposed change (NPF-10-100 and NPF-15-100) implement the requirements for these monitors in Specification 3.3.3.1. The consolidation of the requirements for the NUREG-0737 monitors in Specification 3.3.3.1 will reduce the complexity of the technical specifications while preserving the operability requirements.

SAFETY ANALYSIS

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

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

   Response: No

   The proposed change does not alter the technical specification requirements for wide range radiation monitoring instrumentation. It supports a proposed change to consolidate radiation monitoring instrumentation operability requirements in Specification 3.3.3.1. The wide range radiation monitoring instrumentation affected by the proposed change is not credited in the mitigation of any previously evaluated accident. Therefore, the proposed change does not affect the probability or consequences of any previously evaluated accident.

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

   Response: No

   As stated above, the proposed change supports editorial consolidation of radiation monitoring requirements within the technical specifications. The proposed change does not affect the configuration or operation of the plant. It therefore 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 the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change is editorial in nature supporting the consolidation of technical specification radiation monitoring instrumentation requirements. The consequences of any previously evaluated accident remain unaffected by the proposed change. Therefore, no margin of safety is reduced.

The proposed consolidation of technical specification radiation monitoring instrumentation requirements is similar to example (i) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it is essentially administrative in nature.

SAFETY AND SIGNIFICANT HAZARDS DETERMINATION

Based on the 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"

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one or more radiation monitoring alarm channels inoperable, take the ACTION shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

*See Special Test Exception 3.10.5.

ISSUED TO A CONTROLLED LOCATION

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure - Narrow Range	2	1	20, 21
2. Containment Pressure - Wide Range	2	1	20, 21
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	2	1	20, 21
4. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1	20, 21
5. Pressurizer Pressure - Wide Range	2	1	20, 21
6. Pressurizer Water Level	2	1	20, 21
7. Steam Line Pressure	2/steam generator	1/steam generator	20, 21
8. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator	20, 21
9. Refueling Water Storage Tank Water Level	2	1	20, 21
10. Auxiliary Feedwater Flow Rate	1/steam generator	N.A.	20
11. Reactor Coolant System Subcooling Margin Monitor	2	1	20, 21
12. Safety Valve Position Indicator	1/valve	N.A.	20
13. Spray System Pressure	2	1	20, 21
14. LPSI Header Temperature	2	1	20, 21
15. Containment Temperature	2	1	20, 21
16. Containment Water Level - Narrow Range	2	1	20, 21

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

	INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
17.	Containment Water Level - Wide Range	2	1	20, 21
18.	Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21
19.	Containment Area Radiation - High Range	2	1	20, 21
20.	Main Steam Line Area Radiation	1/steam line	N.A.	20
21.	Condenser Evacuation System Radiation Monitor - Wide Range	1	N.A.	20
22.	Purge/Vent Stack Radiation Monitor - Wide Range*	2	1	22
23.	Cold Leg HPSI Flow	1/cold leg	N.A.	20
24.	Hot Leg HPSI Flow	1/hot leg	N.A.	20

NOTES:

*The two required channels are the Unit 2 monitor and the Unit 3 monitor.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 20 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 21 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 22 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Narrow Range	M	R
2. Containment Pressure - Wide Range	M	R
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
4. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	M	R
5. Pressurizer Pressure (Wide Range)	M	R
6. Pressurizer Water Level	M	R
7. Steam Line Pressure	M	R
8. Steam Generator Water Level (Wide Range)	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Auxiliary Feedwater Flow Rate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. Safety Valve Position Indicator	M	R
13. Spray System Pressure	M	R
14. LPSI Header Temperature	M	R
15. Containment Temperature	M	R
16. Containment Water Level (Narrow Range)	M	R
17. Containment Water Level (Wide Range)	M	R
18. Core Exit Thermocouples	M	R

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS (CONTINUED)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
19. Containment Area Radiation - High Range	(a)	(a)
20. Main Steam Line Area Radiation	(a)	(a)
21. Condenser Evacuation System Radiation Monitor - Wide Range	M	R
22. Purge/Vent Stack Radiation Monitor - Wide Range	M	R
23. Cold Leg HPSI Flow	M	R
24. Hot Leg HPSI Flow	M	R

NOTES:

(a) In accordance with Table 4.3-3.

ATTACHMENT "B"

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one or more accident monitoring channels inoperable, take the ACTION shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

*See Special Test Exception 3.10.5.

ISSUED TO A CONTROLLED LOCATION

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure - Narrow Range	2	1	20, 21
2. Containment Pressure - Wide Range	2	1	20, 21
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	2	1	20, 21
4. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1	20, 21
5. Pressurizer Pressure - Wide Range	2	1	20, 21
6. Pressurizer Water Level	2	1	20, 21
7. Steam Line Pressure	2/steam generator	1/steam generator	20, 21
8. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator	20, 21
9. Refueling Water Storage Tank Water Level	2	1	20, 21
10. Auxiliary Feedwater Flow Rate	1/steam generator	N.A.	20
11. Reactor Coolant System Subcooling Margin Monitor	2	1	20, 21
12. Safety Valve Position Indicator	1/valve	N.A.	20
13. Spray System Pressure	2	1	20, 21
14. LPSI Header Temperature	2	1	20, 21
15. Containment Temperature	2	1	20, 21
16. Containment Water Level - Narrow Range	2	1	20, 21

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TABLE 3.3-

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT		REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
17.	Containment Water Level - Wide Range	2	1	20, 21
18.	Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21
19.	Cold Leg HPSI Flow	1/cold leg	N.A.	20
20.	Hot Leg HPSI Flow	1/hot leg	N.A.	20

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 20 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 21 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Narrow Range	M	R
2. Containment Pressure - Wide Range	M	R
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
4. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	M	R
5. Pressurizer Pressure (Wide Range)	M	R
6. Pressurizer Water Level	M	R
7. Steam Line Pressure	M	R
8. Steam Generator Water Level (Wide Range)	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Auxiliary Feedwater Flow Rate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. Safety Valve Position Indicator	M	R
13. Spray System Pressure	M	R
14. LPSI Header Temperature	M	R
15. Containment Temperature	M	R
16. Containment Water Level (Narrow Range)	M	R
17. Containment Water Level (Wide Range)	M	R
18. Core Exit Thermocouples	M	R

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS (CONTINUED)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
19. Cold Leg HPSI Flow	M	R
20. Hot Leg HPSI Flow	M	R

ATTACHMENT "C"

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

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TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure - Narrow Range	2	1	20, 21
2. Containment Pressure - Wide Range	2	1	20, 21
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	2	1	20, 21
4. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1	20, 21
5. Pressurizer Pressure - Wide Range	2	1	20, 21
6. Pressurizer Water Level	2	1	20, 21
7. Steam Line Pressure	2/steam generator	1/steam generator	20, 21
8. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator	20, 21
9. Refueling Water Storage Tank Water Level	2	1	20, 21
10. Auxiliary Feedwater Flow Rate	1/steam generator	N.A.	20
11. Reactor Coolant System Subcooling Margin Monitor	2	1	20, 21
12. Safety Valve Position Indicator	1/valve	N.A.	20
13. Spray System Pressure	2	1	20, 21
14. LPSI Header Temperature	2	1	20, 21
15. Containment Temperature	2	1	20, 21
16. Containment Water Level - Narrow Range	2	1	20, 21
17. Containment Water Level - Wide Range	2	1	20, 21
18. Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
19. Containment Area Radiation - High Range	2	1	20, 21
20. Main Steam Line Area Radiation	1/steam line	N.A.	20
21. Condenser Evacuation System Radiation Monitor - Wide Range	1	N.A.	20
22. Purge/Vent Stack Radiation Monitor - Wide Range*	2	1	22
23. Cold Leg HPSI Flow	1/cold leg	N.A.	20
24. Hot Leg HPSI Flow	1/hot leg	N.A.	20

NOTES:

*The two required channels are the Unit 2 monitor and the Unit 3 monitor.

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 20 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 21 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 22 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Narrow Range	M	R
2. Containment Pressure - Wide Range	M	R
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
4. Reactor Coolant Inlet Temperature -T_{Cold} (Wide Range)	M	R
5. Pressurizer Pressure (Wide Range)	M	R
6. Pressurizer Water Level	M	R
7. Steam Line Pressure	M	R
8. Steam Generator Water Level (Wide Range)	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Auxiliary Feedwater Flow Rate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. Safety Valve Position Indicator	M	R
13. Spray System Pressure	M	R
14. LPSI Header Temperature	M	R
15. Containment Temperature	M	R
16. Containment Water Level (Narrow Range)	M	R
17. Containment Water Level (Wide Range)	M	R
18. Core Exit Thermocouples	M	R

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS (CONTINUED)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
19. Containment Area Radiation - High Range	(a)	(a)
20. Main Steam Line Area Radiation	(a)	(a)
21. Condenser Evacuation System Radiation Monitor - Wide Range	M	R
22. Purge/Vent Stack Radiation Monitor - Wide Range	M	R
23. Cold Leg HPSI Flow	M	R
24. Hot Leg HPSI Flow	M	R

NOTES:

(a) In accordance with Table 4.3-3.

ATTACHMENT "D"

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more accident monitoring channels inoperable, take the ACTION shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

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TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure - Narrow Range	2	1	20, 21
2. Containment Pressure - Wide Range	2	1	20, 21
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	2	1	20, 21
4. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1	20, 21
5. Pressurizer Pressure - Wide Range	2	1	20, 21
6. Pressurizer Water Level	2	1	20, 21
7. Steam Line Pressure	2/steam generator	1/steam generator	20, 21
8. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator	20, 21
9. Refueling Water Storage Tank Water Level	2	1	20, 21
10. Auxiliary Feedwater Flow Rate	1/steam generator	N.A.	20
11. Reactor Coolant System Subcooling Margin Monitor	2	1	20, 21
12. Safety Valve Position Indicator	1/valve	N.A.	20
13. Spray System Pressure	2	1	20, 21
14. LPSI Header Temperature	2	1	20, 21
15. Containment Temperature	2	1	20, 21
16. Containment Water Level - Narrow Range	2	1	20, 21
17. Containment Water Level - Wide Range	2	1	20, 21
18. Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
19. Cold Leg HPSI Flow	1/cold leg	N.A.	20
20. Hot Leg HPSI Flow	1/hot leg	N.A.	20

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 20 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 21 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

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TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Narrow Range	M	R
2. Containment Pressure - Wide Range	M	R
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
4. Reactor Coolant Inlet Temperature -T_{Cold} (Wide Range)	M	R
5. Pressurizer Pressure (Wide Range)	M	R
6. Pressurizer Water Level	M	R
7. Steam Line Pressure	M	R
8. Steam Generator Water Level (Wide Range)	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Auxiliary Feedwater Flow Rate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. Safety Valve Position Indicator	M	R
13. Spray System Pressure	M	R
14. LPSI Header Temperature	M	R
15. Containment Temperature	M	R
16. Containment Water Level (Narrow Range)	M	R
17. Containment Water Level (Wide Range)	M	R
18. Core Exit Thermocouples	M	R

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TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS (CONTINUED)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
19. Cold Leg HPSI Flow	M	R
20. Hot Leg HPSI Flow	M	R

DESCRIPTION OF PROPOSED CHANGES
NPF-10-102 AND NPF-15-102
AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.3.3.9, Radioactive Gaseous Effluent Monitoring Instrumentation.

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 is required to clarify the technical specification requirements for radioactive gaseous effluent monitoring instrumentation. The proposed change increases operating flexibility by crediting recent and near future design changes (when they are implemented). It makes editorial changes reflecting changes proposed by the NRC staff in the draft revision of NUREG 0472, Standard Radiological Effluent Technical Specifications. Editorial changes are also made within the individual ACTION statements to ensure consistency with the general ACTION provisions of Specification 3.3.3.9. Cross referencing to other specifications not relating to effluent monitoring and the applicability for condenser evacuation system effluent monitoring are revised to be consistent with actual design and FSAR commitments. Other editorial changes are made to improve consistency with the actual plant configuration and operation.

The following revisions are made to Specification 3/4.3.3.9 and its associated Tables 3.3-13 and 4.3-9:

1. The terminology for flow rate measuring devices for each instrument is revised for clarity.

2. A design change has provided automatic closure of the waste gas decay tank isolation valves by the plant vent stack wide range gaseous effluent monitors (2 or 3 RT-7865-1). Item 1a is revised to take credit for this design feature. The plant vent stacks are the final release points for the waste gas system. As a minimum, the proposed change requires only one of the two wide range plant vent stack monitors to be operable. Each unit's plant vent stack is monitored by the wide range plant vent stack monitor associated with that unit. The Unit 2 and Unit 3 plant vent stacks are fed from a common plenum. Exhaust from one unit's vent stack

is representative of exhaust from the other unit's. Data has been accumulated during plant operation to date to correlate the release from one unit's vent stack to the readings obtained from the other unit's vent stack monitor. This correlation enables a representative determination of releases from one unit's vent stack from readings taken from the other unit's vent stack monitor. A minimum of one plant vent stack wide range monitor will effectively monitor and terminate releases from the Waste Gas Holdup System.

The proposed change deletes 2/3 RT-7814 from satisfying the waste gas holdup system noble gas monitoring requirement (Item 1a). This monitor is no longer required to satisfy this function because: (1) a plant vent stack noble gas monitor is always required to be operable by Item 4a; (2) the plant vent stack monitors all provide automatic termination of waste gas hold up system releases; and (3) the plant vent stacks are the final release point for waste gas holdup system releases.

The statement "otherwise suspend release of radioactive effluents via this pathway" is deleted from ACTION 35 which applies to waste gas holdup system noble gas monitoring, to make it consistent with the other gaseous effluent monitoring ACTIONS. This requirement is implicit with failure to meet any gaseous effluent monitoring LCO and ACTION. Calling it out specifically for ACTION 28 only confuses the action to be taken when other gaseous effluent monitoring LCO's and ACTIONS are not met.

3. ACTION 39 is invoked when one or more channels of waste gas holdup system explosive gas monitoring instrumentation is inoperable. Two hydrogen and two oxygen analyzers are provided. One of each is a continuous analyzer, the others are periodic. The continuous hydrogen and oxygen analyzers monitor the waste gas surge tank (waste gas compressor inlet). The periodic analyzers can be aligned to either the surge tank or the waste gas decay tank of interest. The surge tank is of greater interest from the standpoint of preventing explosive gas mixtures in the decay tanks. Because the decay tanks are operated above atmospheric pressure, thereby preventing oxygen inleakage, an explosive gas mixture cannot exist in the decay tanks unless one existed in the surge tank before compression. ACTION 39 assumes that both sets of analyzers are aligned to the surge tank. Should a continuous analyzer on the surge tank become inoperable ACTION 39 currently does not require alignment of the periodic analyzer to the surge tank. Operation of the waste gas houldup system could continue in compliance with ACTION 39 with the surge tank un-monitored. The proposed revision of ACTION 39 resolves this by requiring the remaining operable analyzer channel to be aligned to the waste gas surge tank.

The operability requirements for hydrogen and oxygen analyzers and ACTION 39 are intended to maintain compliance with Specification 3/4.11.2.5, Explosive Gas Mixture. ACTION 39 requires a plant shutdown should both channels of the hydrogen or the oxygen analyzers are inoperable. This is inconsistent with Specification 3/4.11.2.5 and the

requirements of general ACTION "c" of Specification 3/4.3.3.9 neither of which require a plant shutdown even if an explosive gas mixture exists, since they are both 3.0.3 exempt. Therefore in lieu of plant shutdown when both channels are inoperable, ACTION 39 is revised to require grab samples at least once per four hours with analysis within the next four hours to verify compliance with Specification 3/4.11.2.5 and provide adequate assurance that an explosive gas mixture does not exist.

4. Item 4, Plant Vent Stack, is revised to indicate that either the Unit 2 or the Unit 3 vent stack wide range monitors can be used to quantify total plant vent stack releases. As stated above, the Unit 2 and 3 plant vent stacks are fed from a common plenum. Data has been accumulated during plant operation to date to correlate the release from one unit's plant vent stack to the readings obtained from the other unit's vent stack monitor. This correlation enables a representative determination of total plant vent stack releases based on readings from one unit's plant vent stack monitor.

5. The applicability for Item 6, Condenser Evacuation System Monitor, is revised from "all MODES" to "MODES 1-4 with any main steam isolation valve (MSIV) and/or any main steam isolating valve bypass valve not fully closed". The condenser evaucation system is monitored because it is a potential radioactive gaseous release pathway. Primary to secondary leakage is the only source of gaseous acivity which could be potentially released via this pathway. When the MSIV's and MSIV bypass valves are fully closed, this pathway is isolated from the source and, therefore, is not required to be monitored.

6. Item 5, Containment Purge System, is revised to reflect the future addition of a dedicated purge effluent monitor. Installation of a dedicated purge effluent monitor for each unit is required by Units 2 and 3 License Conditions 2(C)17 and 2(C)15, respectively. Notes have been added to indicate which monitors are credited with providing the Item 5 functions before and after the design changes have been implemented. The plant vent stack wide range monitors (RT-7865-1) may, in the future, be equipped to automatically terminate purge releases from their associated unit. The proposed change would recognize the plant vent stack wide range monitors as satisfying the purge effluent monitoring requirements if this design feature is added. ACTION 38 is also revised to provide additional flexibility. As previous submittals have exemplified, reliance on a single monitor for the purge effluent monitoring function has been operationally limiting. Most recently, proposed change NPF-10-87 was submitted to allow the use of the plant vent stack monitor (RT-7865-1) to be used to monitor purge effluent in the event that the containment airborne monitor (RT-7804-1), which currently serves this function, is inoperable. Discussions with the staff raised concerns with NPF-10-87 relating to the action to be taken if a high radiation alarm is received on the other unit's vent stack monitor. This revision to ACTION 38 addresses this staff concern and supercedes the previous submittal. This use of the plant vent stack monitor is viewed as a temporary backup (unless provided with the ability to automatically terminate containment purge) until the dedicated purge effluent monitor is installed.

7. Table 4.3-9 is revised to reflect that channel checks are not required for iodine and particulate samplers. The iodine and particulate samplers are fixed cannisters which are removed weekly in accordance with Specification 4.11.2.1.2, Table 4.11-2, Item D. Channel checks are not appropriate for these samplers and this requirement is deleted from Items 3, 4 and 5 of Table 4.3-9. The surveillance intervals for Item 5 - Containment Purge Noble Gas Activity monitor are revised to be consistent with the Standard Radiological and Effluent Technical Specifications. The current requirements for channel checks each shift and monthly channel functional checks reflect that the containment purge monitoring function is currently performed by an ESFAS monitor. These requirements will still continue to be applied to the ESFAS monitor by Specification 3.3.2. However, when the dedicated purge effluent monitor is installed the effluent monitoring surveillance requirements will apply to it.

8. Table 3.3-13, ACTIONs 35, 36, 37, 38, 39 and 40 limit the period of their compensatory measures to a fixed number of days. However, this is inconsistent with the general ACTION c of Specification 3.3.3.9 which exempts Specifications 3.0.3, 3.0.4 and 6.9.1.13b. With these exemptions, no additional ACTIONs are prescribed when the time limit expires except as provided by general ACTION b. Appropriately, the compensatory measures provided by the ACTIONs may therefore continue until the full provisions of the LCO again are met. Accordingly, the proposed change removes the time limits from ACTIONs 35, 36, 37, 38, 39 and 40. However, SCE fully intends to maintain radioactive effluent monitoring instrumentation in a high state of availability and considers that the burden of compliance with the ACTION requirements provides sufficient encouragement to restore inoperable instruments to operable status in a timely fashion. It is in SCE's best interests to do so. However, there may be circumstances where it may not be possible to restore an inoperable channel to operable status within 30 days. In the event an instrument remains inoperable for greater than thirty days, the reasons why it was not restored in a timely manner will continue to be reported in the semiannual Radioactive Effluent Release Report as required by general ACTION "b." General ACTION "b" is revised to clarify this requirement.

9. With the required sample and/or process flow instrumentation inoperable, ACTION 36 allows effluent releases to continue provided that flow is estimated at least once per 4 hours. ACTION 36 does not specify means by which flow can be estimated. The proposed change revises ACTION 36 to recognize that system design characteristics may be used to estimate flow. If system design characteristics, which are not subject to rapid change, are used to estimate flow, then it is not necessary to estimate flow at 4 hour intervals. The proposed change revises the interval for flow estimation to at least once per 8 hours.

10. ACTION 37 currently provides for grab samples to be taken at 8 hour intervals. Consistent with NUREG 0472 Draft, Revision 3, this interval is increased to 12 hours.

11. The Table 3.3-13 notes cross reference to other specifications relating to other functional requirements for instruments which also serve functions other than effluent monitoring. These references are removed because the functions referred to are specified elsewhere. Only effluent monitoring requirements should appear in this specification.

These revisions will serve to clarify the technical specification requirements for effluent monitoring instrumentation, increase flexibility and improve consistency with the standard Technical Specifications.

Safety Analysis

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

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

   Response: No

   Description Items 1, 7, 8, 9, 10 and 11 above are editorial in nature and have no impact on the probability or consequences of previously analyzed accidents. The proposed changes described in Items 2 and 4 above recognizes the fact that total effluent releases from Units 2 and 3 plant vent stacks, which are fed from a common plenum, can be quantified by the wide range gaseous effluent monitor on one unit's vent stack. This change is supported by data accumulated in operation to date. While this change represents a reduction requirement for effluent monitoring instrumentation, the affected effluent monitoring instrumentation neither contributes to the occurrence of nor is credited in the mitigation of any previously evaluated accident. Therefore, this change does not affect the probability or consequences of any previously evaluated accident.

   The proposed change to ACTION 39 described in Item 3 above removes the requirement for plant shutdown if more than one channel of waste gas holdup system explosive gas monitoring is inoperable. As a compensatory measure, the revised ACTION 39 requires frequent analysis of grab samples to preclude the existence of potentially explosive gas concentrations. This compensatory measure provides assurance that this proposed change will not result in a significant increase in the probability or consequences of any previously evaluated accident.

   The proposed changes to condenser evacuation system effluent monitoring applicability described in Item 5 above remove the requirement to monitor this release path when it is isolated from its source of potential radioactive gaseous effluent. The proposed change maintains the requirement to monitor this path when it is a potential gaseous radioactive effluent pathway. The proposed change affects only condenser

evacuation system effluent monitoring instrumentation, which does not contribute to the occurrence of nor is credited in the mitigation of any previously evaluated accident. Because of these facts, the proposed change does not affect the probability or consequences of any previously evaluated accident.

Item 6 relates to the containment purge system effluent monitoring. Currently this effluent monitoring function is served by containment airborne monitor RT-7804-1. This monitor also serves the ESFAS CPIS function and the RCS leakage detection function. The technical specifications governing these other functions are unaffected by the proposed change. The CPIS function is credited in the evaluation of the limiting in containment fuel handling accident in MODE 6. Because the proposed change does not affect the CPIS requirements, the probability and consequences of this previously evaluated accident remain unchanged.

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

Items 1, 7, 8, 9, 10 and 11 are editorial in nature. Because they make no changes to the configuration of the plant or its operation, they do not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes described in Items 2 and 4 above require fewer instruments to perform plant vent stack monitoring functions than are presently required. However the requirements to monitor plant vent stack effluents and to automatcially terminate releases from the waste gas holdup system remain unchanged. Because the functional requirements remain unchanged the proposed change does not alter the configuration of the plant or its operation in a manner that creates the possibility of a new or different kind of accident from any previously evaluated.

The proposed change described in Item 3 revises ACTION 39 to allow continued plant operation with more than one waste gas holdup system explosive gas monitoring channel inoperable. The revised ACTION 39 includes compensatory measures to preclude the buildup of explosive gas concentrations. With this compensatory measure, the requirement to monitor the waste gas holdup system remains unchanged. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

As described above, Item 6 will allow the use of plant vent stack monitor (RT-7865-1) in lieu of the instrument currently used to monitor containment purge effluent (containment airborne radiation monitor RT-7804-1) when it is inoperable. This provision would be applicable until the dedicated purge effluent monitor is installed. The plant vent

stack monitor does not currently have provisions for automatic termination of containment purge. However, the restrictions included with the use of the plant vent stack monitor and the other diverse means of isolating containment purge (e.g., other CPIS inputs, CIAS and SIAS) assure the purge effluent monitoring will be essentially equivalent to the current provisions. Because purge effluent monitoring is maintained essentially equivalent to the current provisions, the proposed change does not create the possibility for a new or different kind of accident from any previously evaluated.

Item 5 reduces the applicability of the condenser evacuation system monitor from "all MODES" to "MODES 1-4 with any MSIV or MSIV bypass valve is not fully closed". As stated above, the condenser evacuation system is not a potential release path when the plant is in this configuration. This change does not alter the configuration of the plant nor does it create any new mechanisms for initiation of an accident of a new or different kind 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 clarifies the technical specification requirements for radioactive gaseous effluent monitoring instrumentation. Items 1, 7, 8, 9, 10 and 11 described above are editorial in nature and therefore do not involve a reduction in any safety margin. Although Items 2, 3, 4, 5 and 6 involve some reduction in existing requirements, the provisions and associated compensatory measures contained in the proposed change maintain effluent monitoring functional requirements at a level essentially equivalent to the current provisions. Therefore, no margin of safety is significantly reduced by the proposed change.

48 FR 14864 dated April 6, 1983, provided examples of amendments considered not likely to involve a significant hazards consideration. Items 1, 7, 8, 9, 10 and 11 of the proposed change are similar to Example (i) in that they are editorial in nature. Items 2, 3, 4, 5 and 6 are considered to be most similar to Example (vi) in that they involve some degree of reduction of existing Technical Specification requirements, which may be perceived to involve an insignificant reduction in a margin of safety, but are within the acceptance criteria of the Standard Review Plan.

<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.

PSmith:0244F

ATTACHMENT "A"

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.3-13*

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Additionally, if the inoperable instruments are not returned to OPERABLE status within 30 days, explain the next Semi-annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-9.

*See Special Test Exception 3.10.5

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2/3 RT - 7814 or 2/3 RT - 7808	1	*	35
b. Effluent System Flow Rate Measuring Device	1	*	36
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	2	**	39
b. Oxygen Monitor	2	**	39
3. CONDENSER EVACUATION SYSTEM			
a. Noble Gas Activity Monitor - 2RT - 7818 or 2RT - 7870-1	1	*	37, (a)
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	1	*	36
4. PLANT VENT STACK			
a. Noble Gas Activity Monitor - - 2/3 RT - 7808, or 2RT-7865-1 and 3RT-7865-1	1	*	37, (a)
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36
5. CONTAINMENT PURGE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2RT - 7804-1	1	*	38, (b), (c)
b. Iodine Sampler	1	*	40, (c)
c. Particulate Sampler	1	*	40, (b), (c)
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

a) In accordance with Table 3.3-6 ACTION 19

b) In accordance with the ACTION Requirements of Specification 3.4.5.1 (Modes 1, 2, 3 and 4)

c) In accordance with the ACTION Requirement of Specification 3.9.9 (Mode 6)

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days. With two channels inoperable, be in at least HOT STANDBY within 6 hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

TABLE 3.3-13 (Continued)

TABLE NOTATION

Note 1

From August 6, 1982, through September 6, 1982, containment purge with Noble Gas Activity Monitor 2RT-7804-1 inoperable is permissable for no more than two hours per day provided that:

(1) Vent stack monitor 2RT-7865-1 is OPERABLE and aligned to the purge stack for the duration of the purge.

(2) In the event of a high activity alarm on 2RT-7865-1 during the purge, an operator will (a) suspend containment purge and then (b) realign 2RT-7865-1 to the vent stack.

(3) When purging is completed, 2RT-7865-1 is returned to its normal alignment.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2/3 RT - 7814 or 2/3 RT-7808	P	P	R(3)	Q(1)	*
b. Flow Rate Monitor	P	N.A.	R	Q	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor (continuous)	D	N.A.	Q(4)	M	**
b. Hydrogen Monitor (periodic)	D	N.A.	Q(4)	M	**
c. Oxygen Monitor (continuous)	D	N.A.	Q(5)	M	**
d. Oxygen Monitor (periodic)	D	N.A.	Q(5)	M	**
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor - 2RT - 7818, 2RT - 7870-1	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
4. PLANT VENT STACK					
a. Noble Gas Activity Monitor - 2/3 RT - 7808, or 2RT - 7865-1 and 3RT-7865-1	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. CONTAINMENT PURGE SYSTEM(7)					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2 RT - 7804-1	S	P(6)	R(3)	M(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*

TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:#

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists#:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

(6) Prior to each release and at least once per month.

(7) Surveillance of containment airborne monitor 2RT-7807-2 and its associated sampling media, when required OPERABLE by other Specifications, shall be in accordance with the Surveillance Requirement for Containment Purge Effluent monitoring.

#If the instrument controls are not set in the operate mode, procedures shall call for declaring the channel inoperable.

ATTACHMENT "B"

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.3-13*

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Additionally, if the inoperable instrument(s) remain inoperable for greater than 30 days, explain the next Semi-annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-9.

*See Special Test Exception 3.10.5

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2/3 RT-7808, 2RT-7865-1 or 3RT-7865-1	1	*	35
b. Process Flow Rate Monitoring Device	1	*	36
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	2	**	39
b. Oxygen Monitor	2	**	39
3. CONDENSER EVACUATION SYSTEM			
a. Noble Gas Activity Monitor - 2RT - 7818 or 2RT - 7870-1	1	***	37
b. Iodine Sampler	1	***	40
c. Particulate Sampler	1	***	40
d. Associated Sample Flow Measuring Device	1	***	36
e. Process Flow Rate Monitoring Device	1 (4)	***	36
4. PLANT VENT STACK			
a. Noble Gas Activity Monitor - - 2/3 RT - 7808, or 2RT-7865-1 or 3RT-7865-1	1	*	37
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Associated Sample Flow Measuring Device	1	*	36
e. Process Flow Rate Monitoring Device	1 (5)	*	36
5. CONTAINMENT PURGE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2RT - 7828 or 2 RT-7865-1(1)	1 (2)	*	38
b. Iodine Sampler	1 (2)	*	40
c. Particulate Sampler	1 (2)	*	40
d. Process Flow Rate Monitoring Device	1 (3)	*	36
e. Associated Sample Flow Measuring Device	1 (2)	*	36

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

*** MODES 1-4 with any main steam isolation valve and/or any main steam isolating valve bypass valve not fully closed.

(1) Provided 2RT-7865-1 is equipped to automatically terminate containment purge release.

(2) Prior to completion of DCP53N, Containment Airborne Radiation Monitor 2RT-7804-1 performs the functions of 2RT-7828. 2RT-7804-1 is not equipped to monitor purge flow.

(3) Prior to completion of DCP53N, 2RT-7865-1 may perform this function for minipurge only. Otherwise comply with ACTION 36 if another means of continuously monitoring purge flow is not available.

(4) See attached page.

(5) See attached page.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup;

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 8 hours. System design characteristics may be used to estimate flow.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - See attached page

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided that the remaining OPERABLE channel is aligned to the waste gas surge tank. With two channels inoperable operation of this system may continue provided that grab samples are taken at least once pe 4 hours and analyzed with the following 4 hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway

OR

Prior to completion of DCP53N, and with Plant Vent Stack Monitor 2RT-7865-1 not capable of terminating containment purge release, PURGING may continue using 2RT-7865-1 provided that:

1) Plant Vent Stack Monitor 2RT-7865-1 is aligned to the purge stack for the duration of the purge; and,

2) Plant Vent Stack Monitor 2/3 RT-7808 or 3RT-7865-1 is OPERABLE and aligned to the plant vent stack; and,

3) When PURGING is complete, 2RT-7865-1 is realigned to the plant vent stack; and,

4) In the event of a high activity alarm during the PURGE from any of 2RT-7865-1, 3RT-7865-1 or 2/3 RT-7808, an operator immediately suspends containment PURGING and re-aligns 2RT-7865-1 to the Plant Vent Stack.

Notes Cont'd

4) 2 RT-7818 is not equipped to monitor process flow. If another means of continuously monitoring process flow is not available, then comply with ACTION 36.

5) 2/3 RT-7808 is not equipped to monitor plant vent stack flow. If another means of continuously monitoring plant vent stack flow is not available, then comply with ACTION 36.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2RT-7865-1 or 3RT-7865-1 or 2/3 RT-7808	P	P	R(3)	Q(1)	*
b. Process Flow Monitoring Device	P	N.A.	R	Q	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor(continuous)	D	N.A.	Q(4)	M	**
b. Oxygen Monitor(continuous)	D	N.A.	Q(4)	M	**
c. Hydrogen Monitor(periodic)	D	N.A.	Q(5)	M	**
d. Oxygen Monitor(periodic)	D	N.A.	Q(5)	M	**
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor - 2RT - 7818, 2RT - 7870-1	D	M	R(3)	Q(2)	***
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
d. Associated Sample Flow Measuring Device	D	N.A.	R	Q	***
e. Process Flow Rate Monitoring Device (2RT-7870-1)	D	N.A.	R	Q	***
4. PLANT VENT STACK					
a. Noble Gas Activity Monitor - 2/3 RT - 7808, 2RT - 7865-1 or 3RT-7865-1	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
d. Associated Sample Flow Measuring Device	D	N.A.	R	Q	*
e. Process Flow Rate Monitoring Device	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. CONTAINMENT PURGE SYSTEM(7)					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2RT-7828 or 2RT-7865-1	D	P(6)	R(3)	Q(1)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
d. Process Flow Rate Monitoring Device	D	N.A.	R	Q	*
e. Associated Sample Flow Measuring Device	D	N.A.	R	Q	*

TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

*** MODES 1-4 with any main steam isolation valve and/or any main steam isolating valve bypass valve not fully closed.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:#

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists#:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

(6) Prior to each release and at least once per month.

(7) Prior to completion of DCP53N, these surveillance requirements are to be performed on the instrumentation indicated by Table 3.3-13.

#If the instrument controls are not set in the operate mode, procedures shall call for declaring the channel inoperable.

ATTACHMENT "C"

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.*

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Additionally, if the inoperable instruments are not returned to OPERABLE status within 30 days, explain the next Semi-annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-9.

*Continuous monitoring and sampling of the containment purge exhaust directly from the purge stack shall be provided for the low and high volume (8-inch and 42-inch) containment purge prior to startup following the first refueling outage. Containment airborne monitor 3RT-7804-1 or 3RT-7807-2 and associated sampling media shall perform these functions prior to initial criticality. From initial criticality to the startup following the first refueling outage containment airborne monitor 3RT-7804-1 and associated sampling media shall perform the above required functions.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2/3 RT - 7814 or 2/3 RT - 7808	1	*	35
b. Effluent System Flow Rate Measuring Device	1	*	36
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	2	**	39
b. Oxygen Monitor	2	**	39
3. CONDENSER EVACUATION SYSTEM			
a. Noble Gas Activity Monitor - 3RT - 7818 or 3RT - 7870-1	1	*	37, (a)
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	1	*	36
4. PLANT VENT STACK			
a. Noble Gas Activity Monitor - - 2/3 RT - 7808, or 2RT-7865-1 and 3RT-7865-1	1	*	37, (a)
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36
5. CONTAINMENT PURGE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 3RT - 7804-1	1	*	38, (b),(c)
b. Iodine Sampler	1	*	40, (c)
c. Particulate Sampler	1	*	40, (b), (c)
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36

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TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

a) In accordance with Table 3.3-6 ACTION 19

b) In accordance with the ACTION Requirements of Specification 3.4.5.1 (Modes 1, 2, 3 and 4)

c) In accordance with the ACTION Requirement of Specification 3.9.9 (Mode 6)

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days. With two channels inoperable, be in at least HOT STANDBY within 6 hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2/3 RT - 7814 or 2/3 RT-7808	P	P	R(3)	Q(1)	*
b. Flow Rate Monitor	P	N.A.	R	Q	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor (continuous)	D	N.A.	Q(4)	M	**
b. Hydrogen Monitor (periodic)	D	N.A.	Q(4)	M	**
c. Oxygen Monitor (continuous)	D	N.A.	Q(5)	M	**
d. Oxygen Monitor (periodic)	D	N.A.	Q(5)	M	**
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor - 3RT - 7818, 3RT - 7870-1	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
4. PLANT VENT STACK					
a. Noble Gas Activity Monitor - 2/3 RT - 7808, or 2RT - 7865-1 and 3RT-7865-1	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. CONTAINMENT PURGE SYSTEM(7)					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 3RI - 7804-1	S	P(6)	R(3)	M(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:#

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists#:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

(6) Prior to each release and at least once per month.

(7) Surveillance of containment airborne monitor 3RT-7807-2 and its associated sampling media, when required OPERABLE by other Specifications, shall be in accordance with the Surveillance Requirement for Containment Purge Effluent monitoring.

#If the instrument controls are not set in the operate mode, procedures shall call for declaring the channel inoperable.

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ATTACHMENT "D"

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.*

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Additionally, if the inoperable instrument(s) remain inoperable for greater than 30 days, explain the next Semi-annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-9.

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*Continuous monitoring and sampling of the containment purge exhaust directly from the purge stack shall be provided for the low and high volume (8-inch and 42-inch) containment purge prior to startup following the first refueling outage. Containment airborne monitor 3RT-7804-1 or 3RT-7807-2 and associated sampling media shall perform these functions prior to initial criticality. From initial criticality to the startup following the first refueling outage containment airborne monitor 3RT-7804-1 and associated sampling media shall perform the above required functions.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2/3 RT-7808, 2 RT-7865-1 or 3 RT-7865-1	1	*	35
b. Process Flow Rate Monitoring Device	1	*	36
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	2	**	39
b. Oxygen Monitor	2	**	39
3. CONDENSER EVACUATION SYSTEM			
a. Noble Gas Activity Monitor - 3RT - 7818 or 3RT - 7870-1	1	***	37
b. Iodine Sampler	1	***	40
c. Particulate Sampler	1	***	40
d. Associated Sample Flow Measuring Device	1	***	36
e. Process Flow Rate Monitoring Device	1 (4)	***	36
4. PLANT VENT STACK			
a. Noble Gas Activity Monitor - - 2/3 RT - 7808, 2RT-7865-1 or 3RT-7865-1	1	*	37
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Associated Sample Flow Measuring Device	1	*	36
e. Process Flow Rate Monitoring Device	1 (5)	*	36
5. CONTAINMENT PURGE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release 3RT-7828 or 3RT-7865-1 (1)	1 (2)	*	38
b. Iodine Sampler	1 (2)	*	40
c. Particulate Sampler	1 (2)	*	40
d. Process Flow Rate Monitoring Device	1 (3)	*	36
e. Associated Sample Flow Measuring Device	1 (2)	*	36

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TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

*** MODES 1-4 with any main steam isolation valve and/or any main steam isolating valve bypass valve not fully closed.

(1) Provided 3RT-7865-1 is equipped to automatically terminate containment purge release.

(2) Prior to completion of DCP53N, Containment Airborne Radiation Monitor 3RT-7804-1 performs the functions of 3RT-7828. 3 RT-7804-1 is not equipped to monitor purge flow.

(3) Prior to completion of DCP53N, 3RT-7865-1 may perform this function for minipurge only. Otherwise comply with ACTION 36 if another means of continuously monitoring purge flow is not available.

(4) See attached page.

(5) See attached page.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup;

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 8 hours. System design characteristics may be used to estimate flow.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - See attached page

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided that the remaining OPERABLE channel is aligned to the waste gas surge tank. With two channels inoperable, operation of this system may continue provided that grab samples are taken at least once per 4 hours and analyzed within the following four hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway

OR

Prior to completion of DCP53N, and with Plant Vent Stack Monitor 3RT-7865-1 not capable of terminating containment purge release, PURGING may continue using 3RT-7865-1 provided that:

1) Plant Vent Stack Monitor 3RT-7865-1 is aligned to the purge stack for the duration of the purge; and,

2) Plant Vent Stack Monitor 2/3 RT-7808 or 2RT-7865-1 is OPERABLE and aligned to the plant vent stack; and,

3) When PURGING is complete, 3RT-7865-1 is realigned to the plant vent stack; and,

4) In the event of a high activity alarm during the PURGE from any of 3RT-7865-1, 2RT-7865-1 or 2/3 RT-7808, an operator immediately suspends containment PURGING and re-aligns 3RT-7865-1 to the Plant Vent Stack.

Notes Cont'd

(4) 3 RT-7818 is not equipped to monitor process flow. If another means of continuously monitoring process flow is not available, then comply with ACTION 36.

(5) 2/3 RT-7808 is not equipped to monitor plant vent stack flow. If another means of continously monitoring plant vent stack flow is not available then comply with ACTION 36.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2/3 RT-7808 2RT-7865-1 or 3RT-7865-1	P	P	R(3)	Q(1)	*
b. Process Flow Rate Monitoring Device	P	N.A.	R	Q	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor(continuous)	D	N.A.	Q(4)	M	**
b. Oxygen Monitor(continuous)	D	N.A.	Q(4)	M	**
c. Hydrogen Monitor(periodic)	D	N.A.	Q(5)	M	**
d. Oxygen Monitor(periodic)	D	N.A.	Q(5)	M	**
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor - 3RT - 7818, 3RT - 7870-1	D	M	R(3)	Q(2)	***
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
d. Associated Sample Flow Measuring Device	D	N.A.	R	Q	***
e. Process Flow Rate Monitoring Device (3RT-7870-1)	D	N.A.	R	Q	***
4. PLANT VENT STACK					
a. Noble Gas Activity Monitor - 2/3 RT - 7808, 2RT - 7865-1, or 3RT-7865-1	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
d. Associated Sample Flow Measuring Device	D	N.A.	R	Q	*
e. Process Flow Monitoring Device	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. CONTAINMENT PURGE SYSTEM(7)					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 3RI-7828 or 3RI-7865-1	D	P(6)	R(3)	Q (1)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	N.A.
d. Process Flow Rate Monitoring Device	D	N.A.	R	Q	*
e. Associated Sample Flow Measuring Device	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

*** MODES 1-4 with any main steam isolation valve and/or any main steam isolating valve bypass valve not fully closed.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:#

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists#:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

(6) Prior to each release and at least once per month.

(7) Prior to completion of DCP53N, these surveillance requirements are to be performed on the instruments indicated by Table 3.3-13.

#If the instrument controls are not set in the operate mode, procedures shall call for declaring the channel inoperable.

DESCRIPTION OF PROPOSED CHANGES NPF-10-103 AND NPF-15-103 AND SAFETY ANALYSIS

This is a request to revise Technical Specification Sections 3/4.11, Radioactive Effluents, and 3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM.

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 clarifies the requirements of Technical Specification Sections 3/4.11 and 3/4.12 and incorporates revised wording from NUREG 0472, Draft Revision 3, "Standard Radiological Effluent Technical Specifications (RETS) for Pressurized Water Reactors - September 1982". The proposed change does not result in an increase in any effluent release limit and it is essentially editorial in nature in that it, for the most part, incorporates wording which was provided by the staff, reflecting current staff positions. The changes are as follows:

1. RETS definitions of the terms MEMBER(S) OF THE PUBLIC, SITE BOUNDARY and UNRESTRICTED AREA are incorporated into Technical Specification Section 1.0, Definitions.

2. Surveillance requirements 4.11.1.1.1, 4.11.1.1.2 and 4.11.1.1.3 are revised in accordance with the RETS which simplifies 4.11.1.1.1 and 4.11.1.1.2 and deletes 4.11.1.1.3.

3. Table 4.11-1 Radioactive Liquid Waste Sampling Analysis Program is revised as follows:

   A. Note "a" is revised to incorporate RETS wording.

   B. The reference to the ODCM is removed from Note "d". The ODCM neither describes methods for thoroughly mixing batches of liquid radwaste nor is it required to do so.

   C. Note "f" is revised to reflect that the information required is to be reported in the Semiannual Radiological Effluent Release Report.

   D. Note "#" and "##" are deleted since their applicability has expired.

4. Specification 3.11.1.2 is revised to incorporate RETS wording and to recognize that Units 2&3 have a combined radwaste system. Specified dose limits are revised to reflect combined Units 2&3 values versus the current per reactor unit bases. The phrase "to assure that subsequent releases will be in compliance with Specification 3.11.1.2" are deleted from ACTION "a". This will make the ACTION wording more consistent with the wording of 10CFR 50 Appendix I for the action to be taken when the dose limits are exceeded. The reference to Specification 6.9.1.13b is removed from ACTION "b" because it is redundant to the reporting requirements of ACTION "a". Surveillance requirement 4.11.1.2 is revised to reflect that the required dose calculations are for the current calendar quarter and year.

5. Specification 3.11.1.3 is revised to recognize the combined Units 2&3 radwaste system and to reflect dose limits on a combined Units 2&3 bases rather than per reactor unit. The ACTIONs are revised to incorporate new RETS wording.

6. Specification 3.11.1.4 is revised to include the RETS clarification of which temporary tanks are covered by this Specification.

7. Specification 3.11.2. is revised to incorporate revised RETS wording and to clarify that unrestricted areas are those areas at and beyond the SITE BOUNDARY.

8. Table 4.11-2 Radioactive Gaseous Waste Sampling and Analysis Program is revised as follows:

   A. Note "a" is revised to incorporate RETS wording.

   B. Note "b" currently requires analysis for all gaseous release paths following shutdown, startup and thermal power changes of greater than 15% within one hour. Provided that primary coolant activity remains below the 1.0 micro Ci/gm DOSE EQUIVALENT I-131 limit of Specification 3/4.4.7 there will be no significant increase in gaseous effluent activity to merit the additional analyses currently required.

   C. Note "d" is similarly revised to base the requirement for additional analyses of iodine and particulate sampling media on primary coolant activity. Provided that primary coolant activity is below the 1.0 micro Ci/gm DOSE EQUIVALENT I-131 limit of Specification 3/4.4.7, the additional analysis required by Note "d" is not merited.

9. Specification 3.11.2.2 is revised to:

   A. clarify that "from the site" means to areas at and beyond the SITE BOUNDARY.

   B. specify dose limits on a combined Unit 2&3 bases versus per reactor unit.

C. Revise wording of ACTION "a" to be more consistent with the wording of 10CFR50 Appendix I for actions to be taken when dose limits are exceeded.

D. Clarify calculations of cumulative dose contributions due to noble gases are required by Surveillance Requirements 4.11.2.2.

10. Specification 3.11.2.3 is revised to:

A. Incorporate RETS wording

B. Clarify that "from the site" means to areas at and beyond the SITE BOUNDARY

C. Specify dose limits on a combined Units 2&3 bases

D. Revise wording of ACTION "a" to be more consistent with the wording of 10CFR50 Appendix I

E. Clarify the surveillance requirements in accordance with the revised RETS wording.

11. Specification 3/4.11.2.4 is revised to incorporate new RETS wording and clarify that "from the site" means to areas at and beyond the SITE BOUNDARY.

12. Specification 3/4.11.3 is revised to incorporate new RETS wording.

13. Specification 3/4.12.1 is revised to incorporate new RETS wording. In addition, minor clarifications are made to ACTION "c" to specify broadleaf vegetable samples versus leafy vegetable samples and to require reporting of replacement sample location in the "Annual Radiological Environmental Operating Report" which is more appropriate than the "Semiannual Radioactive Effluent Report" suggested by the RETS. Surveillance requirement 4.12.1 is editorially revised for greater clarity.

14. Table 3.12-1 Radiological Environmental Monitoring Program is revised as follows:

A. Item 3b was transferred directly from RETS which assumes the units discharge liquid effluent into a fresh water body which could be used for drinking water. Consistent with Regulatory Guide 1.109 Appendix A, this requirement does not apply to units discharging to oceans. Item 3b is therefore deleted.

B. The word "leafy" in Item 4b is replaced with "broadleaf".

C. Note "f" is deleted in its entirety. This footnote, transferred directly from RETS, assumes that the units use a fresh water river as the ultimate heat sink. For units on nonflowing bodies of water, this note is meaningless and therefore is deleted.

15. Table 4.12-1, Maximum Values for the Lower Limits of Detection is revised as follows:

    A. Note "a" is revised consistent with the new RETS wording.

    B. Note "b" is deleted since there are no drinking water pathways.

16. Specification 3/4.12.2. Land Use Census is revised as follows:

    A. "Broadleaf" is substituted for "leafy".

    B. ACTION "a" incorporates revised RETS wording but requires reports to be made in the "Annual Radiological Environmental Operating Report" versus the "Semiannual Radioactive Effluent Release Reports".

    C. The unnecessary restriction to conduct the land use census between June 1 and October 1, is deleted from Surveillance Requirement 4.12.2 in that it is inconsistent with the RETS. Revised RETs wording is incorporated.

17. Specification 3/4.12.3, Interlaboratory Comparison Program is revised to delete the reference to the ODCM from the Surveillance requirement. The ODCM does not, nor is required to, address the Interlaboratory Comparison Program.

Safety Analysis

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

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

   Response: No

   The proposed change clarifies requirements relating to radioactive effluents and radiological environmental monitoring and is unrelated to any previously evaluated accident. As such, the proposed change does not affect the probability or consequences of any previously evaluated accident.

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

   Response: No

   The proposed change is essentially editorial in nature and does not alter the configuration of the plant or its operation. 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

   As stated above, the proposed change does not affect any radioactive effluent release limits. The proposed change does involve two reductions in existing requirements. Item 8 of the above description reduces the gaseous effluent analyses requirements for some situations provided certain specified conditions are met. If these conditions are met, then the required sampling is meaningless. No reduction in a margin of safety is involved.

   Item 14 of the above description, removes requirements which were transferred from the Standard Technical Specifications but are only applicable to units which use fresh water rivers as their ultimate heat sink and liquid radwaste discharge path. This revision is editorial in nature and although it involves a reduction in existing requirements, it does not involve a decrease in any margin of safety.

48 FR 14864 dated April 6, 1983, provided examples of amendments not likely to involve a significant hazards consideration. The proposed change, with the exception of Item 8, described above is similar to example (i) in that it is editorial in nature. Because Item 8 involves a reduction in existing Technical Specification requirements, it would likely be considered most similar to example (vi), even though no safety margin is reduced.

Safety and Significant Hazards Determination

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

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Attachment "A"

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-4) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11-1. The results of the previous post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

4.11.1.1.3 The radioactivity concentration of liquids discharged from continuous release points shall be determined by collection and analysis of samples in accordance with Table 4.11-1. The results of the analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[d]	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	5×10^{-7}
1. Primary Plant Makeup Storage Tanks			I-131	1×10^{-6}
2. Radwaste Primary Tanks	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	1×10^{-5}
3. Radwaste Secondary Tanks	P Each Batch	M Composite[b]	H-3	1×10^{-5}
4. Miscellaneous Waste Condensate Monitor Tanks			Gross Alpha	1×10^{-7}
5. Neutralization Sump	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e,#]	D Grab Sample	W Composite[c]	Principal Gamma Emitters[f]	5×10^{-7}
1. Steam Generator Blowdown			I-131	1×10^{-6}
2. Turbine Building Sump	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
3. Miscellaneous Waste Evaporator Condensate*	D Grab Sample	M Composite[c]	H-3	1×10^{-5}
4. Salt Water			Gross Alpha	1×10^{-7}
Discharge From Component Cooling Heat Exchanger	D Grab Sample	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
Steam Generator Blowdown Bypass##**			Fe-55	1x10-6

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TABLE 4.11-1 (Continued)

TABLE NOTATION

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

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

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

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

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

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

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TABLE 4.11-1 (Continued)

TABLE NOTATION

b. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed, by a method described in the ODCM, to assure representative sampling.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of system that has an input flow during the continuous release.

f. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

* Sampling of this flow is not required if, at least once per 31 days, condensate monitor tank bypass valve, SA 1415-2½"-200, is verified locked shut.

\# Administrative controls shall provide for composite sampling of the continuous releases per note b vice note c until January 1, 1983. Continuous proportional sampling shall be in accordance with note c from January 1, 1983 and all times subsequent as required by Table 4.11-1.

\## Administrative controls shall provide for composite sampling of the continuous releases per note b vice note c until January 1, 1984. Continuous proportional sampling shall be in accordance with note c from January 1, 1984 and all times subsequent as required by Table 4.11-1.

** Sampling of this flow is not required if at least once per 31 days blowdown bypass isolation valve (S21301MU618 for Steam Generator E088 and S21301MU619 for Steam Generator E089) is verified locked shut.

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LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to an individual from radioactive materials in liquid effluents released, from each reactor unit, from the site (see Figure 5.1-4) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.1.2

b. The provisions of specifications 3.0.3, 3.0.4 and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

LIQUID WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the site (see Figure 5.1-4) when averaged over 31 days, would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.*

APPLICABILITY: At all times.

ACTION:

a. With the liquid radwaste treatment system inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.1.3.2 The liquid radwaste treatment system shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 15 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquid effluents during the previous 92 days.

*Per reactor unit

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outside temporary tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any outside temporary tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each outside temporary tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

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RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

RADIOACTIVE EFFLUENTS

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND TRITIUM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from tritium, radioiodines and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,

b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of tritium, radioiodines, and radioactive materials in particulate form, with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions taken to reduce releases and the proposed actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.2.3.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of the GASEOUS RADWASTE TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the site (see Figure 5.1-3), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Figure 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.*

APPLICABILITY: At all times.

ACTION:

a. With the GASEOUS RADWASTE TREATMENT SYSTEM and/or the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days or with gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

    1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.2.4.2 The GASEOUS RADWASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the GASEOUS RADWASTE TREATMENT SYSTEM equipment and VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 15 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

*These doses are per reactor unit.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within one hour and less than or equal to 2% by volume within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 134,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be OPERABLE and used, as applicable in accordance with a PROCESS CONTROL PROGRAM, for the SOLIDIFICATION and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.

APPLICABILITY: At all times.*

ACTION:

a. With the packaging requirements of 10 CFR Part 20 and/or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 days, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status,

   3. A description of the alternative used for SOLIDIFICATION and packaging of radioactive wastes, and

   4. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

*See Specification 6.13.1.

SURVEILLANCE REQUIREMENTS (Continued)

4.11.3.2 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, other than dewatered bead type, evaporator bottoms, boric acid solutions, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The dose or dose commitment to any member of the public, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which shall be limited to less than or equal to 75 mrem) over 12 consecutive months.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days, which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.11.4. This Special Report shall include an analysis which estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources (including all effluent pathways and direct radiation) for a 12 consecutive month period that includes the release(s) covered by this report. If the estimated dose(s) exceeds the limits of Specification 3.11.4, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190 and including the specified information of § 190.11(b). Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the requirements for dose limitation of 10 CFR Part 20, as addressed in other sections of this technical specification.

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

SURVEILLANCE REQUIREMENTS

4.11.4 Dose Calculations Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.9.1.13. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With fresh leafy vegetable samples or fleshy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause of the unavailability of samples and identifies locations for obtaining replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1, provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations.

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

RADIOLOGICAL ENVIRONMENTAL MONITORING

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
1. AIRBORNE Radioiodine and Particulates	Samples from at least 5 locations 3 samples from offsite locations (in different sectors) of the highest calculated annual average groundlevel D/Q. 1 sample from the vicinity of a community having the highest calculated annual average groundlevel D/Q. 1 sample from a control location 15-30 km (10-20 miles) distant and in the least prevalent wind direction[c]	Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days.[d]	Radioiodine cartridge. Analyze at least once per 7 days for I-131. Particulate sampler. Analyze for gross beta radioactivity $\geq$ 24 hours following filter change. Perform gamma isotopic[b] analysis on each sample when gross beta activity is > 10 times the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
2. DIRECT RADIATION[e]	At least 30 locations including an inner ring of stations in the general area of the site boundary and an outer ring approximately in the 4 to 5 mile range from the site with a station in each sector of each ring. The balance of the stations are in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations.	At least once per 92 days.	Gamma dose. At least once per 92 days.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
3. WATERBORNE			
a. Ocean	4 Locations	At least once per month and composited[f] quarterly	Gamma isotopic analysis of each monthly sample. Tritium analysis of composite sample at least once per 92 days.
b. Drinking	2 Locations	Monthly at each location.	Gamma isotopic and tritium analyses of each sample.
c. Sediment from Shoreline	4 Locations	At least once per 184 days.	Gamma isotopic analysis of each sample.
d. Ocean Bottom Sediments	5 Locations	At least once per 184 days.	Gamma isotopic analysis of each sample.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
4. INGESTION			
a. Nonmigratory Marine Animals	3 Locations	One sample in season, or at least once per 184 days if not seasonal. One sample of each of the following species: 1. Fish-2 adult species such as perch or sheepshead. 2. Crustaceae-such as crab or lobster. 3. Mollusks-such as limpets or seahares.	Gamma isotopic analysis on edible portions.
b. Local Crops	2 Locations	Representative vegetables, normally 1 leafy and 1 fleshy collected at harvest time. At least 2 vegetables collected semiannually from each location.	Gamma isotopic analysis on edible portions semiannually and I-131 analysis for leafy crops.

TABLE 3.12-1 (Continued)

TABLE NOTATION

a. Sample locations are indicated in the ODCM

b. Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

c. The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites which provide valid background data may be substituted.

d. Canisters for the collection of radioiodine in air are subject to channeling. These devices should be carefully checked before operation in the field or several should be mounted in series to prevent loss of iodine.

e. Regulatory Guide 4.13 provides minimum acceptable performance criteria for thermoluminescence dosimetry (TLD) systems used for environmental monitoring. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposed of this table, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges should not be used for measuring direct radiation.

f. Composite samples should be collected with equipment (or equivalent) which is capable of collecting an aliquot at time intervals which are very short (e.g., hourly) relative to the compositing period (e.g., monthly).

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Marine Animals (pCi/Kg, wet)	Local Crops (pCi/Kg, wet)
H-3	2×10^4 [a]			
Mn-54	1×10^3		3×10^4	
Fe-59	4×10^2		1×10^4	
Co-58	1×10^3		3×10^4	
Co-60	3×10^2		1×10^4	
Zn-65	3×10^2		2×10^4	
Zr-Nb-95	4×10^2			
I-131	2	0.9		1×10^2
Cs-134	30	10	1×10^3	1×10^3
Cs-137	50	20	2×10^3	2×10^3
Ba-La-140	2×10^2			

(a) For drinking water samples. This is 40 CFR Part 141 value.

TABLE 4.12-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,c]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m³)	Marine Animals (pCi/kg, wet)	Local Crops (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	1×10^{-2}			
H-3	2000				
Mn-54	15		130		
Fe-59	30		260		
Co-58, 60	15		130		
Zn-65	30		260		
Zr-95	30				
Nb-95	15				
I-131	1[b]	7×10^{-2}		60	
Cs-134	15	5×10^{-2}	130	60	150
Cs-137	18	6×10^{-2}	150	80	180
Ba-140	60				
La-140	15				

TABLE 4.12-1 (Continued)

TABLE NOTATION

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

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

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

Where:

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

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

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

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

2.22 is the number of transformation per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

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

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt shall be used in the calculations.

In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt should be used in the calculation.

TABLE 4.12-1 (Continued)

TABLE NOTATION

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

b. LLD for drinking water.

c. Other peaks which are measurable and identifiable, together with the radionuclides in Table 4.12-1, shall be identified and reported.

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

(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles. For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three miles.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of any other report required by Specification 6.9.1., prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location(s).

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment via the same exposure pathway 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment via the same exposure pathway may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per 12 months between the dates of June 1 and October 1 using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities.

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*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the ODCM shall be included in the Annual Radiological Environmental Operating Report.

Attachment "B"

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-4) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11-1. The results of the previous post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

4.11.1.1.3 The radioactivity concentration of liquids discharged from continuous release points shall be determined by collection and analysis of samples in accordance with Table 4.11-1. The results of the analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

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TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (µCi/ml)[a]
A. Batch Waste Release Tanks[d]	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	5×10^{-7}
1. Primary Plant Makeup Storage Tanks			I-131	1×10^{-6}
2. Radwaste Primary Tanks	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	1×10^{-5}
3. Radwaste Secondary Tanks	P Each Batch	M Composite[b]	H-3	1×10^{-5}
4. Miscellaneous Waste Condensate Monitor Tanks			Gross Alpha	1×10^{-7}
5. Neutralization Sump	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e,#]	D Grab Sample	W Composite[c]	Principal Gamma Emitters[f]	5×10^{-7}
1. Steam Generator Blowdown			I-131	1×10^{-6}
2. Turbine Building Sump	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
3. Miscellaneous Waste Evaporator Condensate*	D Grab Sample	M Composite[c]	H-3	1×10^{-5}
4. Salt Water Discharge From Component Cooling Heat Exchanger			Gross Alpha	1×10^{-7}
Steam Generator Blowdown Bypass##**	D Grab Sample	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.11-1 (Continued)

TABLE NOTATION

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

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

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

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

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

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

TABLE 4.11-1 (Continued)

TABLE NOTATION

b. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed, by a method described in the ODCM, to assure representative sampling.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of system that has an input flow during the continuous release.

f. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

* Sampling of this flow is not required if, at least once per 31 days, condensate monitor tank bypass valve, SA 1415-2½"-200, is verified locked shut.

\# Administrative controls shall provide for composite sampling of the continuous releases per note b vice note c until January 1, 1983. Continuous proportional sampling shall be in accordance with note c from January 1, 1983 and all times subsequent as required by Table 4.11-1.

\## Administrative controls shall provide for composite sampling of the continuous releases per note b vice note c until January 1, 1984. Continuous proportional sampling shall be in accordance with note c from January 1, 1984 and all times subsequent as required by Table 4.11-1.

** Sampling of this flow is not required if at least once per 31 days blowdown bypass isolation valve (S31301MU618 for Steam Generator E088 and S31301MU619 for Steam Generator E089) is verified locked shut.

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RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to an individual from radioactive materials in liquid effluents released, from each reactor unit, from the site (see Figure 5.1-4) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.1.2

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

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RADIOACTIVE EFFLUENTS

LIQUID WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the site (see Figure 5.1-4) when averaged over 31 days, would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.*

APPLICABILITY: At all times.

ACTION:

a. With the liquid radwaste treatment system inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.1.3.2 The liquid radwaste treatment system shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 15 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquid effluents during the previous 92 days.

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*Per reactor unit

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LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outside temporary tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any outside temporary tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each outside temporary tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

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RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

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

TABLE 4.11-2 (Continued)

TABLE NOTATION

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

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

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

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

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

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

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within one hour and less than or equal to 2% by volume within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

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RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 134,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

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RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be OPERABLE and used, as applicable in accordance with a PROCESS CONTROL PROGRAM, for the SOLIDIFICATION and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.

APPLICABILITY: At all times.*

ACTION:

a. With the packaging requirements of 10 CFR Part 20 and/or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 days, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status,

   3. A description of the alternative used for SOLIDIFICATION and packaging of radioactive wastes, and

   4. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

*See Specification 6.13.1.

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SURVEILLANCE REQUIREMENTS (Continued)

4.11.3.2 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, other than dewatered bead type, evaporator bottoms, boric acid solutions, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

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E 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

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

TABLE 4.11-2 (Continued)

TABLE NOTATION

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

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

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

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

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

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

TABLE 4.11-2 (Continued)

TABLE NOTATION

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

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

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

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

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

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

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.2.2.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The dose or dose commitment to any member of the public, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which shall be limited to less than or equal to 75 mrem) over 12 consecutive months.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days, which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.11.4. This Special Report shall include an analysis which estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources (including all effluent pathways and direct radiation) for a 12 consecutive month period that includes the release(s) covered by this report. If the estimated dose(s) exceeds the limits of Specification 3.11.4, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190 and including the specified information of § 190.11(b). Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the requirements for dose limitation of 10 CFR Part 20, as addressed in other sections of this technical specification.

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

SURVEILLANCE REQUIREMENTS

4.11.4 Dose Calculations Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the ODCM.

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.9.1.13. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With fresh leafy vegetable samples or fleshy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause of the unavailability of samples and identifies locations for obtaining replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1, provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations.

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

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TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
1. AIRBORNE Radioiodine and Particulates	Samples from at least 5 locations 3 samples from offsite locations (in different sectors) of the highest calculated annual average groundlevel D/Q. 1 sample from the vicinity of a community having the highest calculated annual average groundlevel D/Q. 1 sample from a control location 15-30 km (10-20 miles) distant and in the least prevalent wind direction[c]	Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days.[d]	Radioiodine cartridge. Analyze at least once per 7 days for I-131. Particulate sampler. Analyze for gross beta radioactivity ≥ 24 hours following filter change. Perform gamma isotopic[b] analysis on each sample when gross beta activity is > 10 times the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
2. DIRECT RADIATION[e]	At least 30 locations including an inner ring of stations in the general area of the site boundary and an outer ring approximately in the 4 to 5 mile range from the site with a station in each sector of each ring. The balance of the stations are in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations.	At least once per 92 days.	Gamma dose. At least once per 92 days.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
3. WATERBORNE			
a. Ocean	4 Locations	At least once per month and composited[f] quarterly	Gamma isotopic analysis of each monthly sample. Tritium analysis of composite sample at least once per 92 days.
b. Drinking	2 Locations	Monthly at each location.	Gamma isotopic and tritium analyses of each sample.
c. Sediment from Shoreline	4 Locations	At least once per 184 days.	Gamma isotopic analysis of each sample.
d. Ocean Bottom Sediments	5 Locations	At least once per 184 days.	Gamma isotopic analysis of each sample.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
4. INGESTION			
a. Nonmigratory Marine Animals	3 Locations	One sample in season, or at least once per 184 days if not seasonal. One sample of each of the following species: 1. Fish-2 adult species such as perch or sheepshead. 2. Crustaceae-such as crab or lobster. 3. Mollusks-such as limpets or seahares.	Gamma isotopic analysis on edible portions.
b. Local Crops	2 Locations	Representative vegetables, normally 1 leafy and 1 fleshy collected at harvest time. At least 2 vegetables collected semiannually from each location.	Gamma isotopic analysis on edible portions semiannually and I-131 analysis for leafy crops.

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TABLE 3.12-1 (Continued)

TABLE NOTATION

a. Sample locations are indicated in the ODCM

b. Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

c. The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites which provide valid background data may be substituted.

d. Canisters for the collection of radioiodine in air are subject to channeling. These devices should be carefully checked before operation in the field or several should be mounted in series to prevent loss of iodine.

e. Regulatory Guide 4.13 provides minimum acceptable performance criteria for thermoluminescence dosimetry (TLD) systems used for environmental monitoring. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposed of this table, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges should not be used for measuring direct radiation.

f. Composite samples should be collected with equipment (or equivalent) which is capable of collecting an aliquot at time intervals which are very short (e.g., hourly) relative to the compositing period (e.g., monthly).

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Marine Animals (pCi/Kg, wet)	Local Crops (pCi/Kg, wet)
H-3	2×10^4 [(a)]			
Mn-54	1×10^3		3×10^4	
Fe-59	4×10^2		1×10^4	
Co-58	1×10^3		3×10^4	
Co-60	3×10^2		1×10^4	
Zn-65	3×10^2		2×10^4	
Zr-Nb-95	4×10^2			
I-131	2	0.9		1×10^2
Cs-134	30	10	1×10^3	1×10^3
Cs-137	50	20	2×10^3	2×10^3
Ba-La-140	2×10^2			

(a) For drinking water samples. This is 40 CFR Part 141 value.

TABLE 4.12-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,c]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Marine Animals (pCi/kg, wet)	Local Crops (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	1×10^{-2}			
H-3	2000				
Mn-54	15		130		
Fe-59	30		260		
Co-58, 60	15		130		
Zn-65	30		260		
Zr-95	30				
Nb-95	15				
I-131	1[b]	7×10^{-2}		60	
Cs-134	15	5×10^{-2}	130	60	150
Cs-137	18	6×10^{-2}	150	80	180
Ba-140	60				
La-140	15				

TABLE 4.12-1 (Continued)

TABLE NOTATION

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

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

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

Where:

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

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

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

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

2.22 is the number of transformation per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

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

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt shall be used in the calculations.

In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background should include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt should be used in the calculation.

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TABLE 4.12-1 (Continued)

TABLE NOTATION

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

b. LLD for drinking water.

c. Other peaks which are measurable and identifiable, together with the radionuclides in Table 4.12-1, shall be identified and reported.

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

(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report ARH-2537 (June 22, 1972).

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles. For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three miles.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of any other report required by Specification 6.9.1., prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location(s).

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment via the same exposure pathway 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment via the same exposure pathway may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per 12 months between the dates of June 1 and October 1 using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities.

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*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

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RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the ODCM shall be included in the Annual Radiological Environmental Operating Report.

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Attachment "C"

The following definitions are added to Section 1.0:

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include nonemployees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall not include nonemployees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the site boundary, access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the site boundary used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-4) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[d]	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	5×10^{-7}
1. Primary Plant Makeup Storage Tanks			I-131	1×10^{-6}
2. Radwaste Primary Tanks	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	1×10^{-5}
3. Radwaste Secondary Tanks	P Each Batch	M Composite[b]	H-3	1×10^{-5}
4. Miscellaneous Waste Condensate Monitor Tanks			Gross Alpha	1×10^{-7}
5. Neutralization Sump	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e]	D Grab Sample	W Composite[c]	Principal Gamma Emitters[f]	5×10^{-7}
1. Steam Generator Blowdown			I-131	1×10^{-6}
2. Turbine Building Sump	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
3. Miscellaneous Waste Evaporator Condensate*	D Grab Sample	M Composite[c]	H-3	1×10^{-5}
4. Salt Water Discharge From Component Cooling Heat Exchanger			Gross Alpha	1×10^{-7}
	D Grab Sample	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
Steam Generator Blowdown Bypass **			Fe-55	1x10-6

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TABLE NOTATION

a. The LLD is defined, for the purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability and only 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

Δt is the elapsed time between midpoint of sample collection and time of counting.

Typical values of E, V, Y and Δt should be used in the calculation.

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

TABLE 4.11-1 (Continued)

TABLE NOTATION

b. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of system that has an input flow during the continuous release.

f. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported in the Semiannual Effluent Release Report Pursuant to Specification 6.9.1.8.

* Sampling of this flow is not required if, at least once per 31 days, condensate monitor tank bypass valve, SA 1415-2 1/2"-200, is verified locked shut.

** Sampling of this flow is not required if at least once per 31 days, blowdown bypass isolation valve (S21301MU618 for Steam Generator E088 and S21301MU619 for Steam Generator E089) is verified locked shut.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from Units 2&3 combined (see Figure 5.1-4) shall be limited:

a. During any calendar quarter to less than or equal to 3.0 mrem to the total body and to less than or equal to 10 mrem to any organ, and

b. During any calendar year to less than or equal to 6 mrem to the total body and to less than or equal to 20 mrem to any organ.

APPLICABILITY: At all times.

ACTION

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and proposed actions to be taken.

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

LIQUID WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE, and the appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from Units 2&3 combined (see Figure 5.1-4) when averaged over 31 days, would exceed 0.12 mrem to the total body or 0.4 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the liquid radwaste treatment system not in operation, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

    1. Explanation of why liquid radwaste was being discharged without treatment, identification of the inoperable equipment or subsystems and the reason for inoperability,

    2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

    3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases shall be projected at least once per 31 days, in accordance with the ODCM.

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outside temporary tank* shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any outside temporary tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each outside temporary tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

*Tanks included in this Specification are those outdoor tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents and that do not have tank overflow and surrounding area drains connected to the liquid radwaste treatment system.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

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

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

b. For iodine - 131, tritium and for all radioactive materials in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.11.2.1.2 The dose rate due to iodine - 131, tritium and radioactive materials in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and anlysis program specificed in Table 4.11-2.

T[illegible]4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

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

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined, for purposes of these Specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background that will be detected with a 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

Typical values of E, V, Y and Δt should be used in the calculation.

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

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Analyses shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period whenever primary coolant activity is equal to or greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131.

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

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10. The requirement for sampling once per 24 hours for at least 7 days need not be performed if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant is less than 1.0 microcurie/gram.

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

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

g. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135 and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported in the "Semiannual Effluent Release Report" pursuant to Specification 6.9.1.8.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from Units 2 and 3 combined to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation and,

b. During any calendar year: Less than or equal to 20 mrad for gamma radiation and less than or equal to 40 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce releases and the proposed corrective actions to be taken.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - IODINE - 131, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from tritium, iodine - 131 and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released, from Units 2&3 combined to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) shall be limited to the following:

A. During any calendary quarter: Less than or equal to 15 mrem to any organ and,

B. During any calendar year: Less than or equal to 30 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

A. With the calculated dose from the release of tritium, iodine - 131, and radioactive materials in particulate form, with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions taken to reduce releases and proposed actions to be taken.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, tritium, and radionuclides in particulate form with half lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases from Units 2 and 3 combined to areas at and beyond the SITE BOUNDARY (see Figure 5.1-3) would exceed either:

a. 0.4 mrad to air from gamma radiation, or

b. 0.8 mrad to air from beta radiation, or

c. 0.6 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Explanation of why gaseous radwaste was being discharged without treatment, identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the methodology and parameters in the ODCM.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within one hour and less than or equal to 2% by volume within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 134,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be SOLIDIFIED or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.*

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures and/or the solid waste system as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, (1) test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements, and (2) take appropriate administrative action to prevent recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM.

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICTION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

*See Specification 6.13.1.

4.11.3.2 Deleted

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specification 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

c. With fresh broad leaf vegetable samples or fleshy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of any other report required by Specification 6.9.1, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Annual Radiological Environmental Operating Report.

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the table and from figures in the ODCM and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities of Table 4.12-1.

SURVEILLANCE REQUIREMENTS

4.2.3 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

c. Using the incore detectors at least once per 31 EFPD's to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

d. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

DESCRIPTION OF PROPOSED CHANGES NPF-10-110 AND NPF-15-110 AND SAFETY ANALYSIS

This is a request to revise Section 4.3.3.2.a of Appendix A, Technical Specifications, for San Onofre Nuclear Generating Station, Units 2 and 3.

INCORE DETECTORS

Existing Specifications:

Units 2 and 3: See Attachment "A"

Proposed Specifications:

Units 2 and 3: See Attachment "B"

Proposed Change:

Change Section 4.3.3.2.a from "By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when..." to "By performance of a CHANNEL CHECK within 24 hours prior to its use if 7 or more days have elapsed since the previous check and at least once per 7 days thereafter when..."

Reason for Proposed Change:

The plant computer, and thus the incore detectors, are often removed from service several times a week. This has no effect on the operability of the incore detector subsystem upon return to service and a CHANNEL CHECK should not be required under these circumstances. The present wording of the technical specifications, however, could be misinterpreted to imply a CHANNEL CHECK is required after such a temporary removal of the plant computer from service.

Safety Analysis:

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

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

   Response: No

Safety Analysis (continued):

This change serves to clarify the existing surveillance requirement to ensure unnecessary CHANNEL CHECKS are not performed for a temporary shutdown of the plant computer for a time duration less than that required for the surveillance (i.e., 7 days). All previous requirements of the surveillance requirement are retained, therefore, there is no significant 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

   Since the requirements and intent of the surveillance remain unchanged (i.e., a CHANNEL CHECK performed at least once each 7 days), no new or different kind of accident possibility is created.

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

   Response: No

   Since the requirements and intent of the surveillance remain unchanged, no significant reduction in a margin of safety is involved.

Safety and Significant Hazards Determination:

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

ATTACHMENT A

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

ATTACHMENT B

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

APPLICABILITY: When the incore detection system is used for monitoring:

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use if 7 or more days have elapsed since the previous check and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB Margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

DESCRIPTION OF PROPOSED CHANGES
NPF-10-131 AND NPF-15-131 AND
SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.3.3.8, RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION.

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 is required to clarify technical specification requirements for radioactive effluent monitoring instrumentation. The proposed change implements editorial changes from NUREG 0472, Draft Revision 3, "Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors - September 1982," improves consistency within the ACTION statements, and allows the use of pumps other than the circulating water pumps to provide dilution to meet the site radioactive effluent concentration limits.

The following changes are made to Specification 3/4.3.3.8:

1. Surveillance Requirement 4.3.3.8.2 is revised to allow the use of any pumps capable of providing adequate dilution in lieu of only the circulating water pumps. The saltwater cooling pumps which provide cooling water to the component cooling water system heat exchangers are also capable of providing dilution for radioactive liquid effluents. The existing surveillance requirement does not specifically allow this.

2. ACTIONs 28, 29, 30, and 31 of Table 3.3-12 are revised to be consistent with the general ACTIONs of Specification 3.3.3.8. The existing ACTIONs 28 through 31 have time limits associated with their compensatory measures. However, the provisions of Specifications 3.0.3, 3.0.4 and 6.9.1.13b are exempted by general ACTION 'a'. This is inconsistent with the time limits on the individual ACTION since no additional action is required. The time limits are therefore deleted.

However, SCE fully intends to maintain radioactive effluent monitoring instrumentation in a high state of availability and considers that the burden of compliance with the ACTION requirements provides sufficient encouragement to restore inoperable instruments to operable status in a timely fashion. It is in SCE's best interests to do so. However, there may be circumstances where it may not be possible to restore an inoperable channel to operable status within 30 days.

In the event an instrument remains inoperable for greater than thirty days, the reasons why it was not restored in a timely manner will continue to be reported in the Semiannual Radioactive Effluent Release Report as required by general ACTION "b". General ACTION "b" is revised to clarify this requirement.

3. The statement "Otherwise suspend release of radioactive effluents via this pathway" is deleted from ACTION 28 to make it consistent with the other liquid effluent monitoring ACTIONs. This requirement is implicit with failure to meet any liquid effluent monitoring LCO and ACTION. Calling it out specifically for ACTION 28 only confuses the action to be taken when other liquid effluent monitoring LCO's and ACTIONs are not met.

4. ACTIONs 29 and 30 are modified to add additional flexibility provided by the design of the plants. Rather than meeting the existing ACTION requirements and continue releases via the affected pathway, the revised ACTIONs would permit isolation of the pathway and diversion of the effluent to the liquid radwaste treatment system for processing as liquid radwaste. This will allow continued operation of the affected system while still meeting liquid effluent monitoring requirements.

SAFETY ANALYSIS

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

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

   Response: **No**

   The proposed change clarifies the requirements for radioactive liquid effluent instrumentation. Item 3 of the proposed change allows the use of pumps other than the circulating water pumps for effluent dilution. This change does not affect the allowable effluent concentration limits. It merely allows the use of other means of providing dilution when required.

   Item 4 of the proposed change recognizes in the liquid effluent monitoring ACTION statements additional flexibility provided by the design of the plant. It does not affect the allowable effluent concentration limits.

The proposed change relates only to radioactive liquid effluent monitoring instrumentation which is not credited in any previously evaluated accident. Therefore, the proposed change does not increase the probability or consequences of any previously evaluated accident.

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

   Response: No

   The proposed change does not alter the configuration of the plant or its operation. The provisions to allow the use of pumps other than the circulating water pumps for effluent dilution and the additional exceptions for ACTIONs 29 and 30 will allow use of existing design features. Therefore, the proposed change 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 the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change is essentially editorial in nature and does not affect any effluent release limit. The additional flexibility provided by the proposed change takes advantage of existing design features. No functional requirements are reduced by the proposed change. Therefore, no margin of safety is reduced.

The proposed revision of radioactive liquid effluent monitoring instrumentation requirements is similar to example (i) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864, dated April 6, 1983, in that it is essentially administrative in nature.

SAFETY AND SIGNIFICANT HAZARDS DETERMINATION

Based on the 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

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.*

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Additionally, if the inoperable instruments are not returned to OPERABLE status within 30 days, explain in the next Semi-annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-8.

4.3.3.8.2 At least once per 4 hours at least one circulating water pump shall be determined to be operating and providing dilution to the discharge structure whenever dilution is required to meet the site radioactive effluent concentration limits of Specification 3.11.1.1.

*See Special Test Exception 3.10.5.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line - 2/3 RT - 7813	1	28
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line - 2 RT - 7817	1	29
c. Turbine Building Sumps Effluent Line - 2 RT - 7821	1	30
d. Steam Generator (E088) Blowdown Bypass Effluent Line - 2RI6759	1	29
e. Steam Generator (E089) Blowdown Bypass Effluent Line - 2RI6753	1	29
2. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	31
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line	1	31
c. Steam Generator (E088) Blowdown Bypass Effluent Line	1	31
d. Steam Generator (E089) Blowdown Bypass Effluent Line	1	31

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.3, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/gram:

a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram DOSE EQUIVALENT I-131.

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 8 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/ml.

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluents Line - 2/3 RT - 7813	D	P	R(2)	Q(1)
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line - 2RT - 7817	D	M	R(2)	Q(1)
c. Turbine Building Sumps Effluent Line - 2RT - 7821	D	M	R(2)	Q(1)
d. Steam Generator (E088) Blowdown Bypass Effluent Line - 2RT6759	D	M	R(2)	Q(1)
e. Steam Generator (E089) Blowdown Bypass Effluent Line - 2RT6753	D	M	R(2)	Q(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R	Q
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line	D(3)	N.A.	R	Q
c. Steam Generator (E088) Blowdown Bypass Effluent Line	D(3)	N.A.	R	Q
d. Steam Generator (E089) Blowdown Bypass Effluent Line	D(3)	N.A.	R	Q

TABLE 4.3-9 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:*

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

*If the instrument controls are not in the operate mode, procedures shall require that the channel be declared inoperable.

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ATTACHMENT B

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.*

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Additionally, if the inoperable instrument(s) remain inoperable for greater than 30 days, explain in the next Semi-annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-8.

4.3.3.8.2 At least once per 4 hours, all pumps required to be providing dilution to meet the site radioactive effluent concentration limits of specification 3.11.1.1 shall be determined to be operating and providing dilution to the discharge structure.

*See Special Test Exception 3.10.5.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line - 2/3 RT - 7813	1	28
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line - 2 RT - 7817	1	29
c. Turbine Building Sumps Effluent Line - 2 RT - 7821	1	30
d. Steam Generator (E088) Blowdown Bypass Effluent Line - 2RT6759	1	29
e. Steam Generator (E089) Blowdown Bypass Effluent Line - 2RT6753	1	29
2. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	31
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line	1	31
c. Steam Generator (E088) Blowdown Bypass Effluent Line	1	31
d. Steam Generator (E089) Blowdown Bypass Effluent Line	1	31

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.3, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/gram:

a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram DOSE EQUIVALENT I-131 OR

c. Lock closed valve HV-3773 and divert flow to T-064 for processing as liquid radwaste.

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 8 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/ml or lock closed valve S22U19-MU077 or S22U19-MU078 and divert flow to the radwaste sump for processing as liquid radwaste.

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluents Line - 2/3 RT - 7813	D	P	R(2)	Q(1)
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line - 2RT - 7817	D	M	R(2)	Q(1)
c. Turbine Building Sumps Effluent Line - 2RT - 7821	D	M	R(2)	Q(1)
d. Steam Generator (E088) Blowdown Bypass Effluent Line - 2RT6759	D	M	R(2)	Q(1)
e. Steam Generator (E089) Blowdown Bypass Effluent Line - 2RT6753	D	M	R(2)	Q(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R	Q
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line	D(3)	N.A.	R	Q
c. Steam Generator (E088) Blowdown Bypass Effluent Line	D(3)	N.A.	R	Q
d. Steam Generator (E089) Blowdown Bypass Effluent Line	D(3)	N.A.	R	Q

TABLE 4.3-8 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:*

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

*If the instrument controls are not in the operate mode, procedures shall require that the channel be declared inoperable.

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ATTACHMENT C

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
1. AIRBORNE Radioiodine and Particulates	Samples from at least 5 locations 3 samples from offsite locations (in different sectors) of the highest calculated annual average groundlevel D/Q. 1 sample from the vicinity of a community having the highest calculated annual average groundlevel D/Q. 1 sample from a control location 15-30 km (10-20 miles) distant and in the least prevalent wind direction[c]	Continuous operof sampler with sample collection as required by dust loading but at least once per 7 days.[d]	Radioiodine cartridge. Analyze at least once per 7 days for I-131. Particulate sampler. Analyze for gross beta radioactivity $\geq$ 24 hours following filter change. Perform gamma isotopic[b] analysis on each sample when gross beta activity is > 10 times the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
2. DIRECT RADIATION[e]	At least 30 locations including an inner ring of stations in the general area of the site boundary and an outer ring approximately in the 4 to 5 mile range from the site with a station in each sector of each ring. The balance of the stations are in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations.	At least once per 92 days.	Gamma dose. At least once per 92 days.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
3. WATERBORNE			
a. Ocean	4 Locations	At least once per month and composited quarterly	Gamma isotopic analysis of each monthly sample. Tritium analysis of composite sample at least once per 92 days.
b. Deleted			
c. Sediment from Shoreline	4 Locations	At least once per 184 days.	Gamma isotopic analysis of each sample.
d. Ocean Bottom Sediments	5 Locations	At least once per 184 days.	Gamma isotopic analysis of each sample.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
4. INGESTION			
a. Nonmigratory Marine Animals	3 Locations	One sample in season, or at least once per 184 days if not seasonal. One sample of each of the following species: 1. Fish-2 adult species such as perch or sheepshead. 2. Crustaceae-such as crab or lobster. 3. Mollusks-such as limpets or seahares.	Gamma isotopic analysis on edible portions.
b. Local Crops	2 Locations	Representative vegetables, normally 1 broadleaf and 1 fleshy collected at harvest time. At least 2 vegetables collected semiannually from each location.	Gamma isotopic analysis on edible portions semiannually and I-131 analysis for leafy crops.

TABLE 3.12-[illegible] (Continued)

TABLE NOTATION

a. Sample locations are indicated in the ODCM

b. Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

c. The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites which provide valid background data may be substituted.

d. Canisters for the collection of radioiodine in air are subject to channeling. These devices should be carefully checked before operation in the field or several should be mounted in series to prevent loss of iodine.

e. Regulatory Guide 4.13 provides minimum acceptable performance criteria for thermoluminescence dosimetry (TLD) systems used for environmental monitoring. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposed of this table, a thermoluminescent dosimeter may be considered to be one phosphor and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges should not be used for measuring direct radiation.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Marine Animals (pCi/Kg, wet)	Local Crops (pCi/Kg, wet)
H-3	2×10^4 (a)			
Mn-54	1×10^3		3×10^4	
Fe-59	4×10^2		1×10^4	
Co-58	1×10^3		3×10^4	
Co-60	3×10^2		1×10^4	
Zn-65	3×10^2		2×10^4	
Zr-Nb-95	4×10^2			
I-131	2	0.9		1×10^2
Cs-134	30	10	1×10^3	1×10^3
Cs-137	50	20	2×10^3	2×10^3
Ba-La-140	2×10^2			

(a) For drinking water samples. This is 40 CFR Part 141 value.

TABLE 4.12-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,c]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Marine Animals (pCi/kg, wet)	Local Crops (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	1×10^{-2}			
H-3	2000				
Mn-54	15		130		
Fe-59	30		260		
Co-58, 60	15		130		
Zn-65	30		260		
Zr-95	30				
Nb-95	15				
I-131	1	7×10^{-2}		60	
Cs-134	15	5×10^{-2}	130	60	150
Cs-137	18	6×10^{-2}	150	80	180
Ba-140	60				
La-140	15				

TABLE 4.12-1 (Continued)

TABLE NOTATION

a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

Where:

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

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

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

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

2.22 is the number of transformation per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

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

Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

Typical values of E, V, Y and Δt shall be used in the calculations.

TABLE 4.12-1 (Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

b. Deleted.

c. Other peaks which are measureable and identifiable, together with the radionuclides in Table 4.12-1, shall be identified and reported.

RADIOACTIVE EFFLUENTS

3/4 12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh broadleaf vegetables in each of the 16 meteorological sectors within a distance of five miles. For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh broadleaf vegetables in each of the 16 meteorological sectors within the distance of three miles.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of any other report required by Specification 6.9.1, identify the new locations in the next Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment via the same exposure pathway 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of any other report required by Specification 6.9.1, identify the new location(s) in the next Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

   The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment via the same exposure pathway may be deleted from this monitoring program after October 31, of the year in which this land use census was conducted.

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per 12 months using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

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*Broadleaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

3/4 12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interloboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report.

TABLE 4.11-2 (Continued)

TABLE NOTATION

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

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

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

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

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

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

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RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.2.2.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

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RADIOACTIVE EFFLUENTS

DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND TRITIUM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from tritium, radioiodines and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,

b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of tritium, radioiodines, and radioactive materials in particulate form, with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions taken to reduce releases and the proposed actions to be taken to assure that subsequent releases will be in compliance with Specification 3.11.2.3.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

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RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of the GASEOUS RADWASTE TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the site (see Figure 5.1-3), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Figure 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.*

APPLICABILITY: At all times.

ACTION:

a. With the GASEOUS RADWASTE TREATMENT SYSTEM and/or the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days or with gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.2.4.2 The GASEOUS RADWASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the GASEOUS RADWASTE TREATMENT SYSTEM equipment and VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 15 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

*These doses are per reactor unit.

NOV 15 1982

DESCRIPTION OF PROPOSED CHANGES NPF-10-109 AND NPF-15-109 AND SAFETY ANALYSIS

This a request to revise Section 4.2.3.c of Appendix A, Technical Specification, for San Onofre Nuclear Generating Station, Units 2 and 3.

POWER DISTRIBUTION LIMITS

Existing Specifications:

Units 2 and 3: See Attachment "A"

Proposed Specifications:

Units 2 and 3: See Attachment "B"

Proposed Change:

Change "...at least once per 31 days..." to read "...at least once per 31 EFPD's..." for Surveillance Requirement 4.2.3.c.

Reason for Proposed Change:

The surveillance is necessitated by physical changes that result from the operation of the reactor and are burnup dependent, therefore, it is appropriate to express the interval in Effective Full Power Days (EFPD's) instead of calendar days. This change will conform with other technical specifications based on fuel exposure.

Safety Analysis:

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

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

   Response: No

   These changes reflect 1) the burnup effect on the incore nuclear instrumentation system which is used to generate Azimuthal Power Tilt in the Core Operating Limit Supervisory System (COLSS) computer program, and 2) the physical changes in the core that result from operation of the reactor. COLSS is used to ensure compliance with the technical specifications but does not perform safety functions, which are provided by the Core Protection Calculators (CPC's). Since the CPC's are by design and intent more conservative than COLSS and since updating Azimuthal Power Tilt

Safety Analysis (continued):

based on reactor operation more accurately reflects its actual change with core operation rather than an arbitrary time requirement, this change does not affect the ability of the plant to protect itself under evaluated accident conditions.

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

   Since the intent and performance of the surveillances remains unchanged except for exchange of a burnup related requirement for a time requirement, which more accurately reflects reactor core conditions, no possibility of a new or different kind of accident is created.

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

   Response: No

   This proposed change does not affect the operational ability of the safety equipment (CPC's) relied upon to perform the safety function and more acccurately reflects the core status for the conservatism built into this function, therefore, this does not involve a reduction in a margin of safety.

Safety and Significant Hazards Determination:

Based on the above discussion, Proposed Changes NPF-10-109 and NPF-15-109 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; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In addition it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the Station on the environment as described in the NRC Final Environmental Statement.

ATTACHMENT A

(Existing Specification)

SURVEILLANCE REQUIREMENTS

4.2.3 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

c. Using the incore detectors at least once per 31 days to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

d. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

ISSUED TO A
CONTROLLED LOCATION

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ATTACHMENT B

(Proposed Specification)

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Additionally, if the inoperable instruments are not returned to OPERABLE status within 30 days, explain in the next Semi-annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-8.

4.3.3.8.2 At least once per 4 hours at least one circulating water pump shall be determined to be operating and providing dilution to the discharge structure whenever dilution is required to meet the site radioactive effluent concentration limits of Specification 3.11.1.1.

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TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line - 2/3 RT - 7813	1	28
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line - 3RT - 7817	1	29
c. Turbine Building Sumps Effluent Line - 3RT - 7821	1	30
d. Steam Generator (E088) Blowdown Bypass Effluent Line - 3RT6759	1	29
e. Steam Generator (E089) Blowdown Bypass Effluent Line - 3RT6753	1	29
2. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	31
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line	1	31
c. Steam Generator (E088) Blowdown Bypass Effluent Line	1	31
d. Steam Generator (E089) Blowdown Bypass Effluent Line	1	31

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.3, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/gram:

a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram DOSE EQUIVALENT I-131.

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 8 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/ml.

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

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TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluents Line - 2/3 RT - 7813	D	P	R(2)	Q(1)
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line - 3RT - 7817	D	M	R(2)	Q(1)
c. Turbine Building Sumps Effluent Line - 3RT - 7821	D	M	R(2)	Q(1)
d. Steam Generator (E088) Blowdown Bypass Effluent Line - 3RT6759	D	M	R(2)	Q(1)
e. Steam Generator (E089) Blowdown Bypass Effluent Line - 3RT6753	D	M	R(2)	Q(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R	Q
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line	D(3)	N.A.	R	Q
c. Steam Generator (E088) Blowdown Bypass Effluent Line	D(3)	N.A.	R	Q
d. Steam Generator (E089) Blowdown Bypass Effluent Line	D(3)	N.A.	R	Q

TABLE 4.3-8 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:*

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

*If the instrument controls are not in the operate mode, procedures shall require that the channel be declared inoperable.

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ATTACHMENT D

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Additionally, if the inoperable instrument (s) remain inoperable for greater than 30 days, explain in the next Semi-annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-8.

4.3.3.8.2 At least once per 4 hours, all pumps required to be providing dilution to meet the site radioactive effluent concentration limits of Specification 3.11.1.1 shall be determined to be operating and providing dilution to the discharge structure.

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TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line - 2/3 RT - 7813	1	28
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line - 3RT - 7817	1	29
c. Turbine Building Sumps Effluent Line - 3RT - 7821	1	30
d. Steam Generator (E088) Blowdown Bypass Effluent Line - 3RT6759	1	29
e. Steam Generator (E089) Blowdown Bypass Effluent Line - 3RT6753	1	29
2. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	31
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line	1	31
c. Steam Generator (E088) Blowdown Bypass Effluent Line	1	31
d. Steam Generator (E089) Blowdown Bypass Effluent Line	1	31

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.3, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/gram:

a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram DOSE EQUIVALENT I-131 OR

c. Lock closed valve HV-3773 and divert flow to T-064 for processing as liquid radwaste

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 8 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/ml OR lock closed valve S22U19-MU077 or S22U19-MU078 and divert flow to the radwaste sump for processing as liquid radwaste.

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

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TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluents Line - 2/3 RT - 7813	D	P	R(2)	Q(1)
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line - 3RT - 7817	D	M	R(2)	Q(1)
c. Turbine Building Sumps Effluent Line - 3RT - 7821	D	M	R(2)	Q(1)
d. Steam Generator (E088) Blowdown Bypass Effluent Line - 3RT6759	D	M	R(2)	Q(1)
e. Steam Generator (E089) Blowdown Bypass Effluent Line - 3RT6753	D	M	R(2)	Q(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R	Q
b. Steam Generator Blowdown (Neutralization Sump) Effluent Line	D(3)	N.A.	R	Q
c. Steam Generator (E088) Blowdown Bypass Effluent Line	D(3)	N.A.	R	Q
d. Steam Generator (E089) Blowdown Bypass Effluent Line	D(3)	N.A.	R	Q

TABLE 4.3-8 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:*

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

*If the instrument controls are not in the operate mode, procedures shall require that the channel be declared inoperable.

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DESCRIPTION OF PROPSOSED CHANGE NPF-15-140 AND SAFETY ANALYSIS

This is a request to revise Section 4.5.2.a of Appendix A, Technical Specifications, for San Onofre Nuclear Generating Station, Unit 3.

ECCS SUBSYSTEMS - $T_{AVG} \geq 350^{o}F$

Existing Specifications:

Unit 3: See Attachment "A"

Proposed Specifications:

Unit 3: See Attachment "B"

Proposed Change:

Delete Manual Isolation Valves 14-081 and 14-082 (items i and j) from Surveillance Specification 4.5.2.a.

Reason of Proposed Change:

These Manual Valves (14-081 and 14-082) do not have remote position indicators and therefore require local verification of valve position. This action requires frequent entry into a contaminated area resulting in a radiation exposure. This area is also considered a confined space and requires oxygen monitoring. Active communication from the area is not available as operators radios cannot transmit from the room. The failure modes and effects analysis in the Final Safety Analysis Report (FSAR) concludes that if 14-081 and 14-082 were inadvertently closed due to operator error no effect on ECCS operation would occur. Since 3HV-8160 and 3HV-8161 are maintained throttled open and open, respectively, and they bypass 14-081 and 14-082, ECCS LPSI flow would not be compromised if 14-081 or 14-082 were closed.

Safety Analysis:

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

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

   Response: No

The closure of 14-081 and 14-082 which are isolation valves for 3HV-0396, which is normally closed, has already been analyzed in the FSAR in the failure modes and effects analysis for the Safety Injection System (FSAR Table 6.3-1 Item 14) which concluded that if 14-081 and 14-082 were inadvertently closed due to operator error, no effect on ECCS operation would occur. Valves 3HV-8160 and 3HV-8161 are maintained throttled open and open, respectively. They bypass 14-081 and 14-082 and form the normal ECCS flowpath. Therefore, ECCS LPSI flow in the Unit 3 as built configuration would not be compromised if 14-081 or 14-082 were closed. Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated?

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

   Response: No

   As noted under Question 1, operation of the Safety Injection System with these valves closed has been analyzed in the FSAR and, due to the normal positions of 3HV-8160 and 3HV-8161, ECCS LPSI flow in the Unit 3 configuration would not be compromised if 14-081 or 14-082 were closed. Therefore, no new or different kinds of accidents from those previously evaluated would be created.

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

   Response: No

   Since the closure of 14-081 and 14-082 for the Safety Injection System is analyzed in the FSAR and since the normally open ECCS flowpath is in parallel to 14-081 and 14-082 which are isolation valves for 3HV-0396, which is normally closed, no significant reduction in a margin of safety is involved.

Safety and Significant Hazards Determination:

Based on the above discussion, Proposed Change NPF-15-140 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; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. 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 signficantly alters the impact of the Station on the environment as described in the NRC Final Environmental Statement.

GvN:1155F

ATTACHMENT A

(Existing Specification)

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 B

(Proposed Specification)

EMERGENCY CORE COOLING SYSTEMS

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.	Deleted		
j.	Deleted		
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.

DESCRIPTION OF PROPOSED CHANGE NPF-10/15-145 AND SAFETY EVALUATION

This is a request to revise Sections 2.C(14) and 2.C(12), Fire Protection, of Facility Operating Licenses NPF-10 and NPF-15, respectively.

Existing License Conditions

Unit 2

(14) Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5)

a. SCE shall maintain in effect and fully implement all provisions of the approved Fire Protection Plan as amended through Amendment 10 and the NRC staff's Fire Protection Review described in the SER, and Supplements 4 and 5 to the SER.

b. Prior to exceeding five (5) percent power, SCE shall install and make operable fixed emergency lighting in access and egress routes to safe shutdown areas. The lighting units shall be sealed-beam units with 8-hour minimum battery power supplies.

c. Prior to exceeding five (5) percent power, SCE shall identify and describe any deviations of the San Onofre 2 fire protection system from the acceptance criteria of Section 9.5.1 of the Standard Review Plan (NUREG-0800, dated July 1981).

d. Prior to fuel loading, SCE shall complete inspection of all Unit 2 and Common Area fire seals, and shall repair deficient seals or implement compensatory measures as defined in the Technical Specifications.

Unit 3

(12) Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5, Section 1.12, SSER #5

a. SCE shall maintain in effect and fully implement all provisions of the approved Fire Protection Plan as amended through Amendment 12 and the NRC staff's Fire Protection Review described in the SER, Supplements 4 and 5 to the SER, and in the Safety Evaluation issued with this license. In

addition, SCE shall meet the technical requirements of Section III.G, "Fire Protection of Safe Shutdown Capability," III.J "Emergency Lighting," and III.O "Oil Collection System for Reactor Coolant Pump" of Appendix R to 10 CFR 50.

b. Prior to exceeding five (5) percent power, SCE shall complete the installation of the following items and shall submit a license amendment request to add items 1, 3, and 5 to the Technical Specifications (Appendix A to this license).

1. Provide fire detection in fire zones 11, 28, 45, 62, and 72.

2. Install metal shrouding on the AFW turbine lube oil system.

3. Install additional sprinklers in the AFW pump room.

4. Provide a 1-1/2 hour fire damper for one of the three charging pump rooms.

5. Provide fire detection and portable extinguishers in the technical support center.

Proposed License Conditions

Unit 2

(14) Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5

SCE shall maintain in effect and fully implement the provisions of the updated Fire Hazards Analysis dated February, 1984 and subsequent revisions and the NRC staff's Fire Protection Review described in the SER and Supplements 4 and 5 to the SER and in the Safety Evaluation issued with Operating License NPF-15 (San Onofre, Unit 3). In addition, SCE shall meet the technical requirements of Section III.G, "Fire Protection of Safe Shutdown Capability," III.J "Emergency Lighting," and III.O "Oil Collection System for Reactor Coolant Pump" of Appendix R to 10 CFR 50 as defined in the updated Fire Hazards Analysis dated February, 1984 and in subsequent revisions.

Unit 3

(12) Fire Protection (Section 9.5.1, SER, SSER #4, SSER #5

SCE shall maintain in effect and fully implement the provisions of the updated Fire Hazards Analysis dated February, 1984 and subsequent revisions and the NRC staff's Fire Protection Review

described in the SER, Supplements 4 and 5 to the SER, and in the Safety Evaluation issued with this license. In addition, SCE shall meet the technical requirements of Section III.G, "Fire Protection of Safe Shutdown Capability," III.J "Emergency Lighting," and III.O "Oil Collection System for Reactor Coolant Pump" of Appendix R to 10 CFR 50 as defined in the updated Fire Hazards Analysis dated February, 1984 and in subsequent revisions.

Reasons for Proposed Change

The purpose of this proposed change is to revise the wording of Fire Protection License Conditions 2.C(14) of SONGS 2 Operating License NPF-10 and 2.C(12) of SONGS 3 Operating License NPF-15 to reflect the fact that the Updated Fire Hazards Analysis (FHA) provides the documentation of the fire protection program for both SONGS 2 and SONGS 3.

SCE has undertaken a systematic review of the total Fire Protection Program for SONGS 2 and 3 to identify and resolve any potential inaccuracies that have not been previously identified with respect to the licensing basis. In addition to formally documenting the licensing basis, SCE's program includes, (1) validation of the FHA to assure that it reflects the appropriate licensing commitments, (2) verification of the accuracy of the FHA with respect to the design documents of SONGS 2 and 3, and (3) walkdowns of the plants to assure implementation in the field of the commitments stated in the FHA. The result of this effort is an Updated FHA that reflects the existing design. The Updated FHA will be submitted to the NRC by separate cover in accordance with the requirements of 10 CFR 50.71(e).

The Updated FHA includes as an appendix, the Design Basis Table for the fire protection program for SONGS 2 and 3. The BTP 9.5-1, Appendix A Design Basis Table documents SONGS 2 and 3 compliance to the requirements and provides justification for any exceptions or alternative compliance noted. The 10 CFR 50, Appendix R Design Basis Table identifies and provides justification for specific known deviations to literal compliance with Appendix R. All but two deviations to Appendix R have already been approved by the NRC in previous licensing documents.

Since the Updated FHA now reflects SCE's conformance to licensing commitments there is no need to duplicate those commitments in the license conditions; reference to the Updated FHA is sufficient. The present wording of the license condition has been the source of past misunderstandings between SCE and Region V with respect to the SONGS 2 and 3 licensing basis. This change should prevent any future misunderstandings and clearly indicates that the Updated FHA is the licensing document that reflects the fire protection licensing basis of SONGS 2 and 3.

In summary, SCE requests the wording of License Conditions 2.C(14) and 2.C(12) be revised to reflect the Updated FHA and to reference the Updated FHA for compliance to 10 CFR 50, Appendix R.

Safety Evaluation

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

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

   Response: No

   The proposed change does not affect the accident analysis or Technical Specification requirements.

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 clarifies SCE's compliance with respect to the NRC's fire protection program requirements for SONGS 2 and 3. This clarification does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No

   As stated above, the proposed change will not involve a significant reduction in a margin of safety.

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

Safety and Significant Hazards Consideration Determination

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

GVanNoordennen:0697F

DESCRIPTION OF PROPOSED CHANGE NPF 10/15-146 AND SAFETY EVALUATION

This is a request to revise the following Technical Specifications of Facility Operating Licenses NPF-10 and NPF-15:

3/4.3.3.7	FIRE DETECTION INSTRUMENTATION
3/4.7.8.2	SPRAY AND/OR SPRINKLER SYSTEMS
3/4.7.8.3	FIRE HOSE STATIONS
3/4.7.9	FIRE RATED ASSEMBLIES
B 3/4.3.3.7	FIRE DETECTION INSTRUMENTATION (BASES)
B 3/4.7.8	FIRE SUPPRESSION SYSTEMS (BASES)
B 3/4.7.9	FIRE RATED ASSEMBLIES (BASES)

Existing Specifications

Unit 2

See Attachment A.

Unit 3

See Attachment B.

Proposed Specifications

Unit 2

See Attachment C.

Unit 3

See Attachment D.

Description of Proposed Change

Fire Detection Instrumentation

Units 2 and 3

Revise Table 3.3-11, in the LIMITING CONDITION FOR OPERATION of Technical Specification 3.3.3.7, to read as shown in Attachment C for Unit 2 and to read as shown in Attachment D for Unit 3.

Revise the ACTION statements of Technical Specification 3.3.3.7 to read as follows:

> With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11:
>
> a. For areas outside containment accessible during plant operation, establish one of the following within 1 hour:

1. The operability of the automatic suppression system in the area; or

2. An hourly fire watch patrol for those areas without suppression.

b. For areas inside containment, inspect containment if accessible during during plant operation at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

c. The provisions of specifications 3.0.3 and 3.0.4 are not applicable.

Revise surveillance specification 4.3.3.7.1 to read as follows:

4.3.3.7.1 a. The above required early warning detectors and the actuating flame detectors which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

b. The above required heat actuating detectors shall be tested as follows: At least one detector per signal string shall be tested every six months. The detector(s) selected for testing shall be rotated so that all detectors in a signal string are tested every five years.

c. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 1 week unless performed in the previous 6 months.

Delete surveillance specification 4.3.3.7.3

Revise and renumber surveillance specification 4.3.3.7.4 to read as follows:

4.3.3.7.3 Following a seismic event (basemat acceleration greater than or equal to 0.05g):

a. Within 2 hours each accessible zone shown in Table 3.3-11 located outside containment shall be inspected for fires, and

b. Within 1 hour monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5 until 4.3.3.7.3.c is completed, or within 8 hours inspect each accessible zone shown in Table 3.3-11 located inside containment, and

c. Within 72 hours perform an engineering evaluation to verify the OPERABILITY of the fire detection system in each zone shown in Table 3.3-11.

Reasons for Proposed Change

The purpose of the proposed changes to Table 3.3-11 is to reflect the information presented in the Updated Fire Hazards Analysis and to reduce the number of detectors required to be operable.

During the revision of the Fire Hazards Analysis for SONGS 2 and 3, a systematic review of the fire detection capability was conducted. The review included: 1) plant walkdowns to verify detection type and spacing, and 2) and analysis of the required spacing based on recommendations found in NFPA 72E and manufacturer's data. Based on this review and analysis the number of detectors required to be operable was reduced when it could be demonstrated that area or hazard coverage was maintained and response time was not lessened significantly.

The proposed Table 3.3-11 was developed using the following process:

1. From the Updated Fire Hazards Analysis a list was developed of all areas/zones containing safe shutdown or safety-related (not required for safe shutdown) equipment.

2. All of the above areas which did not contain detection were eliminated from the list.

3. The areas which remained were evaluated to determine the minimum number of detectors required to ensure adequate coverage and timely response.

4. The results were compared with other plants which have previously received NRC approval for this change.

The following changes resulted from this process:

1. All charcoal filters now require the thermistor strip detection system to be operable. The charcoal filter temperature differential detector system has experienced numerous operational problems and low reliability. It will be removed from service upon NRC approval of this proposed change.

2. All areas which appeared on the current Table 3.3-11 with "none" entered for detectors have been deleted.

3. All areas which do not contain safe shutdown or safety related systems have been deleted.

The purpose of the proposed changes to the ACTION statements is to reflect enhanced monitoring and response capabilities. The following provides the technical bases for the proposed changes:

1. Fire alarms annunciate in the Control Room and are acknowledged by Operations personnel. The fire response organization monitors the fire computer, investigates fire alarms, notifies the Control Room of valid fire annunciation, and initiates compensatory action for impaired fire protection measures. The fire computer is monitored

by an alarm assessor on a 24-hour basis. The alarm assessor's sole function is to respond to trouble and alarm signals indicative of a fire for the fire suppression and detection systems. Therefore, areas with automatic suppression are monitored continuously.

2. SONGS 2 and 3 has a 5-man fire brigade providing coverage on a 24-hour basis. This brigade consists solely of State Certified fire fighters/emergency medical technicians. The brigade is trained in fighting the types of fires encountered in nuclear power plants. The brigade is required to drill quarterly, however, currently they drill weekly to ensure readiness and maintain effectiveness. The brigade has 2 fully equipped fire trucks. The fire brigade brings their own equipment to the scene and are knowledgeable in the location of fire fighting equipment provided throughout the plant.

3. Fire areas containing an automatic suppression system will ensure that the fire is extinguished before the barrier is degraded. For areas without automatic suppression the current hourly fire watch patrol will be maintained.

4. Surveillance of fire detectors has caused station personnel to be unnecessarily exposed to radiation. The addition of the words "accessible during plant operation" will provide station management the perogative to establish priority of radiation and/or other life-threatening safety hazards over fire hazards.

The purpose of the proposed change to Surveillance Specification 4.3.3.7.1 is based upon the detector's excellent performance in service and the testing required by the National Fire Protection Association Code (NFPA) 72E, Chapter 8, Section 3.2.2. The Code requires heat detectors to be tested at least once per five year intervals. To meet this requirement at least one detector per signal string will be tested every six months and the detector(s) selected will be rotated so that all detectors in a signal string are tested every five years. The frequency of the channel functional test required during each cold shutdown has been revised from exceeding 24 hours to exceeding 1 week unless performed in the previous 6 months. The revised testing frequency is consistent with the surveillance described above and minimizes possible delays in resuming unit operation due to completion of the surveillance requirement.

The purpose of deleting surveillance Specification 4.3.3.7.3 is that all fire detection systems in Units 2 and 3 have supervised circuits. Since there are no unsupervised circuits this section is not applicable and should be deleted.

The purpose of the proposed change to Surveillance Specification 4.3.3.7.4 is to reflect the fact that fire zones inside containment may not be accessible within 2 hours of a seismic event. It may be necessary to delay inspection of fire zones inside containment which are temporarily inaccessible due to radiation and/or other life threatening safety hazards. Each temporary inaccessible zone will be inspected for fires as soon as the high radiation or unsafe condition is nullified.

Safety Evaluation

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

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

   Response: No

   The Updated Fire Hazards Analysis, Section 7.0, discusses the consequences of the design basis fire, for each fire area/zone containing safe shutdown systems. This analysis assumed the loss of the unwrapped safe shutdown circuits in the fire areas of concern. Unacceptable consequences were not found. The proposed changes, while reducing the number of detectors required to be operable, do not change the design basis fire; therefore, the changes will not increase the probability or consequences of an accident previously evaluated.

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

   Response: No

   The accidents previously evaluated considered the loss of all unwrapped safe shutdown circuitry within a fire area. The proposed changes, while reducing the number of detectors required to be operable, do not affect the accident analyzed.

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

   Response: No

   SRP 9.5.1, "Fire Protection Program", Section C.5.a (3) states that "Fire detectors should, as a minimum, be selected and installed in accordance with NFPA 72E, "Automatic Fire Detectors." The reduction in the minimum number of detectors required to be operable was determined based on an analysis using the guidelines of NFPA 72E and manufacturer's recommendations (where NFPA 72E refers to the manufacturer's recommendations). Therefore the functionality of the detection system is not degraded to an unacceptable level. Adequate coverage is maintained and timely response is ensured.

The proposed revision is similar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it may reduce in some way a safety margin but where the results of the change are clearly within the acceptable criteria specified in the Standard Review Plan.

Safety and Significant Hazards Consideration Determination

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

Description of Proposed Change

Spray and/or Sprinkler Systems

Units 2 and 3

Revise Table 3.7-5, in the LIMITING CONDITION FOR OPERATION of Technical Specification 3.7.8.2, to read as shown in Attachment C for Unit 2 and to read as shown in Attachment D for Unit 3.

Revise the ACTION statements of Technical Specification 3.7.8.2 to read as follows:

a. With one or more of the required spray and/or sprinkler systems protecting redundant safe shutdown systems outside containment accessible during plant operation, inoperable, establish one of the following within 1 hour:

   1. a) The operability of the detection system in the area; and

      b) Backup fire suppression equipment if applicable;* or

   2. a) An hourly fire watch patrol for those areas without detection; and

      b) Backup fire suppression equipment if applicable.*

b. For other areas outside containment accessible during plant operation, establish the operability of the detection system in the area or an hourly fire watch patrol, within one 1 hour.

* Fire hose will be run within 4 hours of entering the ACTION statement if an operable water supply is not available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Fire hose will be supplied by the fire brigade responding to a fire if an operable water supply is available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Any additional backup fire suppression equipment is provided by the fire brigade responding to a fire.

c. For areas inside containment with inoperable fire detection equipment and with one or more of the above required spray and/or sprinkler systems inoperable, inspect containment if accessible during plant operation at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

d. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

Revise the surveillance requirement of Technical Specification 4.7.8.2.b to read as follows:

b. At least once per 31 days during each COLD SHUTDOWN exceeding 1 week or REFUELING by verifying that each valve (manual, power-operated or automatic) inside containment in the flow path is in its correct position.

Revise the surveillance requirement of Technical Specification 4.7.8.2.d.1 to read as follows:

d. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system except for the manually activated deluge-water spray system for the charcoal filters, and:

a. Verifying that the automatic valves in the flow path actuate to their correct positions on a test signal, and

b. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

Reasons for Proposed Change

The purpose of the proposed change to Table 3.7-5 is to reflect the new fire area/zone designations and incorporate those spray and/or sprinkler systems protecting safe shutdown or safety related equipment not currently in Table 3.7-5, or to delete those spray and/or sprinkler systems currently in Table 3.7-5 which are not protecting safety related or safe shutdown equipment.

The new fire area/zone designations are a result of the revision to the Fire Hazards Analysis. The deletions or additions to the table are a result of the detailed review of safe shutdown/safety related equipment location carried out as part of the FHA revision process.

The process for developing Table 3.7-5 was as follows:

1. From the Updated Fire Hazards Analysis a list was developed of all areas/zones containing safe shutdown or safety related (not required for safe shutdown) equipment.

2. All of the above areas identified in the Updated FHA which did not contain spray and/or sprinkler systems were eliminated from the above list.

3. Those areas which remained were listed in Table 3.7-5.

Four changes resulted from this process:

1. The R. R. Tunnel in the Fuel Handling Building was deleted because no safe shutdown or safety related equipment is located in this area;

2. The truck ramp in the Radwaste Building was deleted because no safe shutdown or safety related equipment is located in this area; and

3. The deluge-water spray system was added to the Auxiliary Feedwater Pump Room to reflect the addition of this system.

4. The charcoal filter water spray systems that are manually actuated protecting safety-related equipment were added.

The purpose of the proposed changes to the ACTION statements is to reflect enhanced detection and response capabilities. The following provides the technical bases for the proposed changes:

1) Fire alarms annunciate in the Control Room and are acknowledged by Operations personnel. The fire response organization monitors the fire computer, investigates fire alarms, notifies the Control Room of valid fire annunciation, and initiates compensatory action for impaired fire protection measures. The fire computer is monitored by an alarm assessor on a 24-hour basis. The alarm assessor's sole function is to respond to trouble and alarm signals indicitive of a fire for the fire detection and suppression system. Therefore, areas with detection are monitored continuously.

2) SONGS 2 and 3 has a 5-man fire brigade providing coverage on a 24-hour basis. This brigade consists solely of State Certified fire fighters/emergency medical technicians. The brigade is trained in fighting the types of fires encountered in nuclear power plants. The brigade is required to drill quarterly, however, currently they drill weekly to ensure readiness and maintain effectiveness. The brigade has 2 fully equipped fire trucks. The fire brigade brings their own equipment to the scene and are knowledgable in the location of fire fighting equipment provided throughout the plant. Therefore, backup fire suppression equipment is provided by the brigade responding to the fire.

3) In fire areas containing an OPERABLE fire detection system a continuous fire watch is not needed to detect a fire and notify the fire brigade. In fire areas containing redundant safe shutdown circuits which could be affected by a single fire, one train is wrapped with a material having a 1-hour rating. Therefore, an hourly fire watch patrol will ensure that the fire is detected and extinguished before the barrier is degraded.

4) The operable detection equipment in containment will provide early warning of any fire in the area. The inspections at 8 hour intervals will allow for the determination of any developing situation which would be controlled by backup fire suppression equipment. Based upon the continuous monitoring by the alarm assessor and the enhanced response capabilities of the fire brigade the potential for damage to safety related and safe shutdown equipment is minimized.

5) Surveillance of sprays and/or sprinkler systems has caused station personnel to be unnecessarily exposed to radiation. The addition of the words "accessible during plant operation" will provide station management the perogative to establish priority of radiation and/or other life-threatening safety hazards over fire hazards.

The requirement for a special report has been deleted from the Technical Specification. Reporting will be done through the LER process.

The frequency for verifying valve positions inside containment has been revised from each cold shutdown to each cold shutdown exceeding one week. The revised verification frequency minimizes possible delays in resuming Unit operation due to completion of the surveillance requirement.

The purpose of the proposed change to surveillance specification 4.7.8.2.d.1 is to exempt the manually activated deluge-water spray systems which protect the charcoal filters. Cycling each valve in the flow path would cause actuation of this system and cause an unnecessary replacement of the charcoal filters. The other portions of Technical Specification 4.7.8.2 will ensure that the spray systems for the charcoal filters are demonstrated OPERABLE.

Safety Evaluation

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

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

   Response: No

   The updated Fire Hazards Analysis, Section 7.0, discusses the consequences of the design basis fire for each fire area/zone containing safe shutdown systems. This analysis assumed the loss of the unwrapped safe shutdown circuits in the fire areas of concern. Unacceptable consequences were not found. The proposed changes do not change the design basis fire; therefore, the changes will not increase the probability or consequences of an accident previously evaluated.

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

   Response: No

   The accidents previously evaluated considered the loss of all unwrapped safe shutdown circuitry within a fire area. The proposed changes, while changing the compensatory measures of the ACTION statement, do not affect the accident analyzed.

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

   Response: No

   The fire suppression system is designed under the guidance of SRP 9.5.1. The SRP does not address the interval in which an inoperable system must be returned to operable status nor does it address the action to be carried out regarding inoperable systems. The continuous monitoring by the alarm assessor and enhanced response capabilities of the station fire brigade ensures that the margin of safety is maintained.

The proposed revision is simillar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it may reduce in some way a margin of safety, but where the results of the change are clearly within acceptable criteria specified in the Standard Review Plan.

Safety and Significant Hazards Consideration Determination

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

Description of Proposed Change

Fire Hose Stations

Units 2 and 3

Revise Table 3.7-6 in the LIMITING CONDITION FOR OPERATION of Technical Specification 3.7.8.3, to read as shown in Attachment C for Unit 2 and to read as shown in Attachment D for Unit 3.

Revise the ACTION statements of Technical Specification 3.7.8.3 to read as follows:

a. With one or more of the above required fire hose stations inoperable, within 4 hours establish backup means of fire suppression if applicable* and within 24 hours post signs above the inoperable hose station(s) and related valves.

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

* Fire hose will be run within 4 hours of entering the ACTION statement if an operable water supply is not available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Fire hose will be supplied by the fire brigade responding to a fire if an operable water supply is available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Any additional backup fire suppression equipment is provided by the fire brigade responding to a fire.

Reasons for Proposed Change

The purpose of the proposed change to Table 3.7-6 is to reflect the new fire area/zone designations and to add fire hose stations deemed necessary to fight a fire in areas containing safe shutdown or safety related systems.

The new fire area/zone designations are a result of the revision to the Fire Hazards Analysis. The addition of the hose stations in certain areas is the result of a systematic evaluation of manual fire suppression capability. This evaluation assumed 60 feet of hose and a 40 foot hose stream.

The following hose stations were added to Table 3.7.6 for Units 2 or 3:

24	Turbine Bldg.
28	A/C Room - Safety Equipment Bldg.
29	A/C Room - Safety Equipment Bldg
31	Piping Room - Safety Equipment Bldg
32	Corridor - Auxiliary Radwaste
33	Corridor - Auxiliary Radwaste
34	Corridor - Auxiliary Radwaste
36	Corridor - Auxiliary Radwaste
37	Corridor - Auxiliary Radwaste
38	Corridor - Auxiliary Radwaste
40	Corridor - Auxiliary Radwaste
41	Corridor - Auxiliary Radwaste
43	Roof - Auxiliary Control
44	Corridor - Auxiliary Radwaste
45	Corridor - Auxiliary Radwaste
46	Corridor - Auxiliary Radwaste
51	Corridor - Auxiliary Radwaste
52	Cable Spreading Rm Corridor - Auxiliary Control
53	Lobby - Auxiliary Radwaste
54	Lobby - Auxiliary Control
55	Corridor - Auxiliary Control
58	Corridor - Auxiliary Control

59	Corridor - Auxiliary Control
63	Office Area - Auxiliary Control
87	Turbine Building
91	A/C Room - Safety Equipment
92	Corridor - Safety Equipment
94	Piping Room - Safety Equipment
95	Intake Structure
96	Intake Structure
97	Diesel Generator
98	Deisel Generator
99	Diesel Generator
100	Diesel Generator
101	Hall-Mezzanine - Auxiliary Control
102	Corridor - Auxiliary Radwaste
103	Corridor - Auxiliary Radwaste
106	Corridor - Auxiliary Radwaste
107	Corridor - Auxiliary Radwaste

The hose stations in the south cable riser gallery and south Cable Spreading Room of the Auxiliary Control Building (hose stations 113, 114, 115, 116 and 117) were deleted from Table 3.7-6 for Unit 2 since they only affect Unit 3 systems. Likewise, hose stations in the north cable riser gallery and north cable spreading room of the Auxiliary Control Building were deleted from Table 3.7-6 for Unit 3 since they only affect Unit 2 systems.

The purpose of the proposed changes to the ACTION statements is to reflect enhanced detection and response capabilities. The following provides the technical bases for the proposed changes:

1) SONGS 2 and 3 has a 5-man fire brigade providing coverage on a 24-hour basis. This brigade consists solely of State Certified fire fighters/emergency medical technicians. The brigade is trained in fighting the types of fires encountered in nuclear power plants. The brigade is required to drill quarterly, however, currently they drill weekly to ensure readiness and maintain effectiveness. The brigade has 2 fully equipped fire trucks. The fire brigade brings their own equipment to the scene and are knowledgable in the location of fire fighting equipment provided throughout the plant. Therefore, backup fire suppression equipment (compensatory hose) is provided by the brigade responding to the fire.

2) The combination of the passive fire protection features and the response capabilities of the fire brigade is adequate to minimize potential damage to safe shutdown or safety related systems.

The requirement for a special report for Unit 2 has been deleted from the Technical Specification. Reporting will be done through the LER process.

Safety Evaluation

1. Will the 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 Updated Fire Hazards Analysis, Section 7.0, discusses the consequences of the design basis fire for each fire area/zone containing safe shutdown equipment. This analysis assumed the loss of the unwrapped safe shutdown circuits in the fire areas of concern. Unacceptable consequences were not found. The proposed changes do not change the design basis fire; therefore, the changes will not increase the probability or consequences of an accident previously evaluated.

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

   Response: No

   The accidents previously evaluated considered the loss of all unwrapped safe shutdown circuitry within the area. The proposed changes, while removing the need for routing compensatory fire hoses to an affected area, do not affect the accident analyzed.

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

   Response: No

   The fire hose stations are designed under the guidance of SRP 9.5.1. The SRP does not address the requirement of laying compensatory hose. The station fire brigade responds to fight fires with its own equipment. Should the need arise, the brigade is trained to lay up to 300 feet of hose quickly and efficiently. A compensatory hose within 300 feet does not increase the margin of safety, therefore removal of that requirement does not involve a significant reduction in the margin of safety.

The proposed revision is similar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it may reduce in some way a safety margin but where the results of the change are clearly within the acceptable criteria specified in the Standard Review Plan.

Safety and Significant Hazards Consideration Determination

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

Description of Proposed Change

Fire Rated Assemblies

Units 2 and 3

Revise the APPLICABILITY statement of Technical Specification 3/4.7.9 to read as follows:

When equipment protected by the fire rated assemblies is required to be OPERABLE.

Revise the ACTION statements of Technical Specification 3.7.9 to read as follows:

a. With one or more of the above required fire rated assemblies and/or sealing devices separating portions of redundant systems important to safe shutdown within a fire area/zone accessible during plant operation, inoperable, within 1 hour either establish the OPERABILITY of the fire detectors or automatic suppression system in the area/zone, or establish a continuous fire watch.

b. With one or more of the above required fire rated assemblies and/or sealing devices separating other fire areas/zones accessible during plant operation inoperable, within 1 hour either establish the OPERABILITY of the fire detectors or automatic suppression system on at least one side of the inoperable assembly, or establish an hourly fire watch patrol.

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

Revise surveillance requirement 4.7.9.1 to read as follows:

4.7.9.1 Each of the above required fire doors accessible during plant operation shall be verified OPERABLE by:

a. Verifying at least once per 24 hours the position of each closed fire door and insure the doorways are free of obstructions.

b. Verifying at least once per 7 days the position of each locked closed fire door.

c. Inspecting at least once per 6 months the release and closing mechanism and latches.

Reasons for Proposed Change

The purpose of the proposed change to the APPLICABILITY statement is to reflect the fact that equipment protected by fire rated assemblies is that which is required to be operable to reach and/or maintain the plant in cold shutdown. The technical specification states that areas separating redundant safe shutdown or safety-related areas shall be protected by fire rated assemblies. The technical specification requirements should therefore be

applicable only to equipment required to be operable to maintain the plant in a safe condition. For example, if the plant is in a cold shutdown condition the only barriers required to be operable would be those protecting equipment required to maintain the plant in a cold shutdown condition.

The purpose of the proposed changes to the ACTION statement is to reflect enhanced detection and response capabilities. The following provides the technical bases for the proposed changes:

1) Fire alarms annunciate in the Control Room and are acknowledged by Operations personnel. The fire response organization monitors the fire computer, investigates fire alarms, notifies the Control Room of valid fire annunciation, and initiates compensatory action for impaired fire protection measures. The fire computer is monitored by an alarm assessor on a 24-hour basis. The alarm assessor's sole function is to respond to trouble and alarm signals indicitive of a fire for the fire suppression and detection systems. Therefore, areas with detection or automatic suppression are monitored continuously.

2) SONGS 2 and 3 has a 5-man fire brigade providing coverage on a 24-hour basis. This brigade consists solely of State Certified fire fighters/emergency medical technicians. The brigade is trained in fighting the types of fires encountered in nuclear power plants. The brigade is required to drill quarterly, however, currently they drill weekly to ensure readiness and maintain effectiveness. The brigade has 2 fully equipped fire trucks. The fire brigade brings their own equipment to the scene and are knowledgable in the location of fire fighting equipment provided throughout the plant. Therefore, upon arrival of the brigade, a fire will be quickly extinguished.

3) For areas/zones without detection and suppression, a distinction in fire watch requirements has been made for the following reasons:

   a. If the inoperable assembly is separating a portion of redundant systems important to safe shutdown within a fire area/zone, a continuous watch must be posted until the assembly is repaired to ensure that safe shutdown function is not degraded.

   b. If the inoperable assembly is separating fire areas or zones containing safe shutdown or safety related systems an hourly fire watch is adequate because boundaries separating fire areas/zones are either of a 3-hr, 2-hr, 1-hr or heavy concrete construction. This substantial construction would preclude the propagation of the design basis fire beyond the area/zone boundaries.

4) Surveillance of fire doors has caused station personnel to be unnecessarily exposed to radiation. The addition of the words "accessible during plant operation" will provide station management the perogative to establish priority of radiation and/or other life-threatening safety hazards over fire hazards.

The combination of enhanced monitoring of the active fire protection systems and response capability of the fire brigade ensures that fire damage will be limited.

The purpose of the proposed changes to the surveillance specification is to reflect the fact that door accessibility has been a problem, to delete the fire door supervisory system requirement and to delete the automatic hold open, release, closing mechanism or latching requirements since those features do not exist on any Unit 2/3 fire doors. The following provides the technical bases for the proposed changes:

1. Surveillance of fire doors has caused station personnel to be unnecessarily exposed to radiation. The addition of the words "accessible during plant operation" will provide station management the perogative to establish priority of radiation and/or other life-threatening safety hazards over fire hazards.

2. Units 2 and 3 do not have a fire door supervisory system. Fire doors which are also security doors are monitored by the security door supervisory system. These fire doors are locked closed and are checked per the requirements of 4.7.9.1.b. Therefore, the channel functional test requirement should be eliminated from Technical Specification 4.7.9.1.

3. Units 2 and 3 do not have any automatic features on any fire doors. The doors will continue to be inspected at least once per 6 months to verify the operability of the release and closing mechanism and latches. The functional test requirement should be eliminated since it only applies to automatic doors.

Safety Evaluation

1. Will the 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 updated Fire Hazards Analysis, Section 7.0, discusses the consequences of the design basis fire for each fire area/zone containing safe shutdown systems. Unacceptable consequences were not found. The proposed changes to the ACTION statements provide an equivalent level of compensatory protection. Therefore the design basis accident is not affected.

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 accidents previously evaluated considered the loss of all unwrapped safe shutdown circuitry within the area. The proposed changes, while changing the compensatory measures of the ACTION statement, do not affect the design basis fire.

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

Response: No

Fire rated assemblies are located taking into consideration the guidance of Standard Review Plan 9.5.1. The SRP does not address compensatory measures to be taken when an assembly is inoperable. The compensatory measures proposed in this change, provide an adequate level of protection, therefore there is not a significant reduction in the margin of safety.

The proposed revision is similar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it may reduce in some way a safety margin but where the results of the change are clearly within the acceptable criteria specified in the Standard Review Plan.

Safety and Significant Hazards Consideration Determination

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

Description of Proposed Change

Fire Detection Instrumentation (Bases)

Units 2 and 3

Revise the second paragraph to read as follows:

> In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols or establishment of the automatic suppression system in the affected areas is required to provide detection or suppression capability until the inoperable instrumentation is restored to OPERABILITY. Areas accessible during plant operation are defined as areas that do not pose radiation and/or life-threatening safety hazards.

Revise the (0.02g) in the third paragraph to (0.05g).

Reason for Proposed Change

The first editorial change defines when an area is accessible during plant operation to perform the required actions or surveillances and takes credit for the automatic suppression system to detect and suppress fires. Life-threatening hazards and radiation hazards warrant a higher priority than a fire hazard.

The second editorial change corrects an inconsistency between the Technical Specifications and the Bases.

Safety Evaluation

1. Will the 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

   This is an editorial change. It does not involve a change to any 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

   This is an editorial change. It does not create a new or different kind of accident.

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

   Response: No

   This is an editorial change. It does not affect a margin of safety.

The proposed revision is similar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it may reduce in some way a safety margin but where the results of the change are clearly within the acceptable criteria specified in the Standard Review Plan.

Safety and Significant Hazards Consideration Determination

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

Description of Proposed Change

Fire Suppression Systems (Bases)

Units 2 and 3

Revise the second paragraph to read as follows:

> In the event that portions of the fire suppression systems are inoperable, alternate fire protection features are required until the operable equipment is restored to service. In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

Add the following sentence to the third paragraph:

> Areas accessible during plant operation are defined as areas that do not pose radiation and/or other life-threatening safety hazards.

Delete the fourth paragraph.

Reasons for Proposed Change

The first and third changes are editorial changes to 1) reflect the proposed changes in the Technical Specification, 2) combine the descriptions of actions required when portions of the fire suppression system are inoperable and, 3) delete the special reporting requirements.

The second change defines when an area is accessible during plant operation to perform the required actions or surveillances. Life-threatening hazards and radiation hazards warrant a higher priority than a fire hazard.

Safety Evaluation

1. Will the 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

   These are editorial changes. They do not involve a change to any 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

   These are editorial changes. They do not create a new or different kind of accident.

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

   Response: No

   These are editorial changes. They do not affect a margin of safety.

The proposed revision is similar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it may reduce in some way a safety margin but where the results of the change are clearly within the acceptable criteria specified in the Standard Review Plan.

Safety and Significant Hazards Consideration Determination

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

Description of Proposed Change

Fire Rated Assemblies (Bases)

Units 2 and 3

Add the following sentence:

Areas accessible during plant operation are defined as areas that do not pose radiation and/or other life-threatening safety hazards.

Reason for Proposed Change

The editorial change defines when an area is accessible during plant operation to perform the required actions or surveillances. Life-threatening hazards and radiation hazards warrant a higher priority than a fire hazard.

Safety Evaluation

1. Will the 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

   This is an editorial change. It does not involve a change to any 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

   This is an editorial change. It does not create a new or different kind of accident.

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

   Response: No

   This is an editorial change. It does not affect a margin of safety.

The proposed revision is similar to example (vi) of amendments not likely to involve a significant hazards consideration published in 48 FR 14864 dated April 6, 1983 in that it may reduce in some way a safety margin but where the results of the change are clearly within the acceptable criteria specified in the Standard Review Plan.

Safety and Significant Hazards Consideration Determination

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

GvN:1005F

ATTACHMENT A

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

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

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.7.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

4.3.3.7.4 Following a seismic event (basemat acceleration greater than or equal to 0.05 g):

a. Within 2 hours each zone shown in Table 3.3-11 shall be inspected for fires, and

b. Within 72 hours an engineering evaluation shall be performed to verify the OPERABILITY of the fire detection system in each zone shown in Table 3.3-11.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
1	Containment						
	Cable Tray Areas Elev 63'3"			10			
	Cable Tray Areas Elev 45'			9			
	Cable Tray Areas Elev 30'			4			
	Elevator Machinery Room			1			
	Combustible Oil Area Two steam generator rooms				32		
	Charcoal Filter Area Elev 45'	2					
2	Penetration Elev 63'6"			12			
4	New Fuel Storage Area and Spent Fuel Pool Areas						
	Spent Fuel Pool		4				
	New Fuel Pool		3				
5	Control Building Elev 70'						
	Cable Riser Gallery Rm 423			2	24		
	Cable Riser Gallery Rm 449			3	24		
6	Control Building Elev 70'						
	Radiation Chemical Lab Rms 421, 420	1					
7	Radwaste Elev 63'6"						
	Chemical Storage Area Rm 503			1			
	Radwaste Control Panel Rm 513			1			
	Storage Area Rm 523			1			
	Hot Machine Shop	1					
8	Radwaste Elev 63'6"						
	Waste Decay Tank Rms 511A		None				
9	Fuel Handling Building Elev 45'						
	Emgy. A.C. Unit Rm 309-Train A	1		1			
	Emgy. A.C. Unit Rm 302-Train B	1		1			
10	Penetration Elev 45'			6			

*The fire detection instruments located within the Containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

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TABLE 3.3-11 (Continued)

Zone	Instrument Location'	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
11	S.E.B. Roof and Main Steam Relief Valves	None					
12	Control Building Elev 50'						
	Cable Riser Gallery Rm 305			3	42		
	Cable Riser Gallery Rm 315			3	40		
13A	Control Building Elev 50' Emgy. HVAC Unit Rm 309A	1					
13B	Control Building Elev 50' Emgy. HVAC Unit Rm 309B	1					
14	Radwaste Elev 24'						
	Boric Acid Makeup Tank Rm 204B	None					
	Boric Acid Makeup Tank Rm 204A	None					
15	Control Building Elev 30'						
	ESF Switchgear Rm 308A			2			
	ESF Switchgear Rm 308B			2			
16	Radwaste Elev 37' & 50' Ion Exchangers	None					
17	Diesel Generator Building						
	Train A			3		4	
	Train B			3		4	
18	Diesel Fuel Oil Storage Tank Underground Vaults	None					
20	Condensate Storage Tank T-121	None					
21	Nuclear Storage Tank T-104	None					
22	Auxiliary Feedwater Pump Room			2		6	
23	Fuel Handling Bldg Elev 30' Spent Fuel Pools Heat Exchange Room 209	None					
28	Penetration Elev. 30'	2					

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TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
29	Control Building Elev 30'						
	Cable Riser Gallery Rm 236			3	51		
	Cable Riser Gallery Rm 224			3	52		
30	Electrical Tunnel Elev 30'6"			13	50		
31	Control Building Elev 30'			29			
32A	Control Building Elev 30'						
	Fan Room Rm 219 & Corridor Rm 221	2		1			
32B	Control Building Elev 30'						
	Fan Room Rm 233 & Corridor Rm 234	2		1			
34	Radwaste Elev 9' & 24'						
	Secondary Radwaste Tank Rms 126A,B & 127A,B	None					
35	Radwaste Elev 9' & 24'						
	Spent Resin Tank Rms 125A,B	None					
36	Fuel Handling Building Elev 17'6"						
	Spent Fuel Pool Pump Rm 107			2			
37	Radwaste Elev 24'						
	Letdown Heat Exchanger Rms 209A,B	None					
38	Radwaste Elev 24'						
	Letdown Control Valve Rms 218A,B	None					
39	Radwaste Elev 24'						
	Filter Crvd Tank Rm 216	None					
40	Radwaste Elev 9' & 24'						
	Primary Radwaste Tank Rms 211A,D	None					
41	Control Building Elev 9'						
	Cable Spreading Rm 111A			17	36		
	Cable Spreading Rm 111B			14	36		
42	Control Building Elev 9'						
	Cable Riser Gallery Rm 110			6	44		
	Cable Riser Gallery Rm 112			6	39		

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
43	Control Building Elev 9'						
	Emgy. Chiller Rm 115			2			
	Emgy. Chiller Rm 117			2			
44	Intake Structure						
	Pump Rm T2-106			4			
	Pump Rm T3-106			4			
45	Penetration Area Elev 9' & 15'						
	Piping Penetration Area 15'	None					
48	Safety Equipment Building 9'						
	CCW HX and Piping Rm 022-025	None					
50	Radwaste Elev 9'						
	Charging Pump Rms 106A-F			6			
51	Radwaste Elev 9'						
	Boric Acid Makeup Tank Rms 105A-D	None					
53	Electrical Tunnel Elev 9'6", 11'6", (-) 2'6"			21	54		
54	Safety Eqpmt Bldg Elev 15'6" & 8'						
	Shutdown HX Rms 003, 004, 016, 018	None					
55	Safety Eqpmt Bldg Elev 8'						
	Chemical Storage Tank Rm 019			1			
56	Safety Eqpmt Bldg Elev 8'						
	Component Cooling Water Surge Tank Rms 020, 021	None					
57	Safety Eqpmt Bldg Elev 15'6"						
	Pump Rm 005			1			
58	Radwaste Elev 37'						
	Reactor Trip System Rms 308A-D, 309-A-C			9			
59	Safety Eqpmt Bldg Elev 15'6"						
	Pump Rm 001			1			

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
60	Safety Eqpmt Bldg Elev 15'6" Pump Rm 015			1			
61	Safety Eqpmt Bldg Elev 15'6" Component Cooling Water Pump Rms 006, 007, 008			3			
62	Radwaste Elev 50' Volume Control Valve Rooms	None					
63	Control Building Elev 50' Corridor			12			
64	Control Building Elev 50' Vital Power Distribution Rms 310A-H			8			
65	Control Building Elev 50' Battery Rms 306B-J			8			
66	Control Building Elev 50' Evacuation Rm 311			1			
67	Radwaste Elev 63'6" Cable Riser Gallery Rm 506A Cable Riser Gallery Rm 506B			2 2	4 4		
68	Penetration 9' - 63'6" Cable Riser Shaft			1	21		
69	Safety Eqpmt Bldg Elev 5'3" Salt Water Cooling Piping Rm 010	None					
70	Radwaste Elev 24' Duct Shaft Rms 222A,B	None					
72	Control Building Elev 70' Corridor 401	None					
75	Refueling Water Storage Tank T-005	None					
76	Refueling Water Storage Tank T-006	None					

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	FLAME	SMOKE	Actuation HEAT	FLAME	SMOKE
78	Control Building Elev 9'						
	Corridor Rm 105			4			
79	Control Building Elev 50'						
	ESF Switchgear Rm 302A			2			
	ESF Switchgear Rm 302B			2			
80	Radwaste Elev 37' & 50'						
	Duct Snaft Rms	None					
81	Radwaste Elev 63'6"						
	Duct Shaft Rms 527A,B	None					
83	Salt Water Cooling Tunnel			6*			
84	Safety Eqpmt Bldg Elev 8'						
	HVAC Rm 017			3			

*3 in UNIT 2, 3 in UNIT 3

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.8.2 The spray and/or sprinkler systems listed in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas outside containment in which redundant systems or components could be damaged; for other areas outside containment, establish an hourly fire watch patrol.

b. With one or more of the above required spray and/or sprinkler systems inside containment inoperable, restore the system to OPERABLE status within 24 hours or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 7 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.8.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) outside of containment in the flow path is in its correct position.

b. At least once per 31 days during each COLD SHUTDOWN or REFUELING by verifying that each valve (manual, power operated or automatic) inside containment in the flow path is in its correct position,

c. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a test signal, and

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray and wet pipe spray sprinkler headers to verify their integrity, and

3. By a visual inspection of each spray/sprinkler head to verify the spray pattern is not obstructed.

e. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

TABLE 3.7-5

Safety Related Spray and/or Sprinkler Systems

Hazard	Location	No. of Systems	System Type
Reactor Coolant Pumps	Containment	4	Deluge-Water Spray
R.R. Tunnel	Fuel Hand. Bldg.	1	Wet Pipe
Truck Ramp	Radwaste Bldg.	1	Wet Pipe
Cable Tunnel	Section 1	1	Deluge-Water Spray
Cable Tunnel	Section 2	1	Deluge-Water Spray
Cable Tunnel	Section 3	1	Deluge-Water Spray
Cable Tunnel	Section 4	1	Deluge-Water Spray
Cable Tunnel	Section 5	1	Deluge-Water Spray
Cable Tunnel	Section 6	1	Deluge-Water Spray
Cable Tunnel	Section 7	1	Deluge-Water Spray
Cable Tunnel	Section 8	1	Deluge-Water Spray
Cable Tunnel	Section 9	1	Deluge-Water Spray
Cable Tunnel	Section 10	1	Deluge-Water Spray
Cable Tunnel Riser	Fuel Hand. Bldg.	1	Deluge-Water Spray
Cable Gallery	Radwaste Bldg.	2*	Deluge-Water Spray
Cable Risers El. 9 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 30 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 50 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 70 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Spreading Room	Control Bldg.	4*	Deluge-Water Spray
Emergency A.C. Unit - Train A	Fuel Handling Bldg.	1**	Deluge-Water Spray
Emergency A.C. Unit - Train B	Fuel Handling Bldg.	1**	Deluge-Water Spray
Diesel Generator	DG Building	2	Pre-action Sprinkler
HVAC Room 309A; Corridor 303	Control Bldg. 50'	1	Wet Pipe
Auxiliary Feedwater Pump Room	Tank Bldg. 30'	1	Pre-action Sprinkler
Fan Room 233 and Corridor 234	Control Bldg. 30'	1	Wet Pipe
Salt Water Cooling Pumps and Salt Water Cooling Tunnel	Intake Structure	1	Wet Pipe
CCW Heat Exchangers and Piping Room; A/C Room 017	Safety Equipment Bldg.	1	Wet Pipe
Corridor 401	Control Bldg. 70'	1	Wet Pipe
Corridor 105	Control Bldg. 9'	1	Wet Pipe

*One half of these systems are designated Unit 3, but are required to be OPERABLE for Unit 2 operation.
**Charcoal filter deluge systems are manually actuated.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.8.3 The fire hose stations shown in Table 3.7-6 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-6 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppresion; otherwise route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the station to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.8.3 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.

   2. Removing the hose for inspection and re-racking, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

TABLE 3.7-6

FIRE HOSE STATIONS

LOCATION	ELEVATION	STATION NUMBER
Containment Bldg. - Unit 2	63'-6"	130
Containment Bldg. - Unit 2	63'-6"	1
Containment Bldg. - Unit 2	63'-6"	8
Containment Bldg. - Unit 2	45'-0"	2
Containment Bldg. - Unit 2	45'-0"	5
Containment Bldg. - Unit 2	45'-0"	9
Containment Bldg. - Unit 2	30'-0"	3
Containment Bldg. - Unit 2	30'-0"	6
Containment Bldg. - Unit 2	30'-0"	10
Containment Bldg. - Unit 2	17'-6"	4
Containment Bldg. - Unit 2	17'-6"	7
Containment Bldg. - Unit 2	17'-6"	11
Electrical Penetration Area - Unit 2	45'-0"	120
Electrical Penetration Area - Unit 2	45'-0"	121
Electrical Penetration Area - Unit 2	63'-6"	122
Electrical Penetration Area - Unit 2	63'-6"	123
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	9'-0"	109
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	9'-0"	114
Cable Spreading Room-Auxiliary Bldg. Control Area	9'-0"	108
Cable Spreading Room-Auxiliary Bldg. Control Area	9'-0"	113
Cable Spreading Room Corridor-Auxiliary Bldg. Control Area	9'-0"	48
Cable Spreading Room Corridor-Auxiliary Bldg. Control Area	9'-0"	60
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	30'-0"	110
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	30'-0"	115
Corridor (North)-Auxiliary Bldg. Control Area	30'-0"	49
Corridor (South)-Auxiliary Bldg. Control Area	30'-0"	61
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	50'-0"	111
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	50'-0"	116
Corridor (North)-Auxiliary Bldg. Control Area	50'-0"	50
Corridor (South)-Auxiliary Bldg. Control Area	50'-0"	62
HVAC Room Corridor-Auxiliary Bldg. Control Area	50'-0"	56
HVAC Room Corridor-Auxiliary Bldg. Control Area	50'-0"	57
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	70'-0"	112
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	70'-0"	117
Fuel Handling Bldg.-Unit 2	63'-6"	118
Fuel Handling Bldg.-Unit 2	63'-6"	119

3/4.7.9 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.9 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, ventilation duct, and piping penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.9.1 Each of the above required fire doors shall be verified OPERABLE by:

a. Verifying at least once per 24 hours the position of each closed fire door and that doors with automatic hold-open and release mechanisms are free of obstructions.

b. Verifying at least once per 7 days the position of each locked closed fire door.

c. Performing a CHANNEL FUNCTIONAL TEST at least once per 31 days of the fire door supervision system.

d. Inspecting at least once per 6 months the automatic hold-open, release and closing mechanism and latches.

e. Performing a functional test at least once per 18 months of automatic hold open, release, closing mechanisms and latches.

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SURVEILLANCE REQUIREMENTS

4.7.9.2 At least once per 18 months the above required fire rated assemblies and penetration sealing devices other than fire doors shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/and associated hardware.

c. Performing a visual inspection of at least 10% of each type (mechanical and electrical) of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

The OPERABILITY of the remote shutdown instrumentation in Panel L411 ensures that sufficient capability is available to permit shutdown and maintenance of COLD SHUTDOWN of the facility in the event of a fire in the cable spreading room, control room or remote shutdown panel, L042.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

Since the fire detectors are non-seismic, a plant visual inspection for fires is required within two hours following an earthquake ($\geq$ 0.02g). Since safe shutdown systems are protected by seismic Category I barriers rated at two and three hours, any fire after an earthquake should be detected by this inspection before safe shutdown systems would be affected. Additionally, to verify the continued OPERABILITY of fire detection systems after an earthquake, an engineering evaluation of the fire detection instrumentation in the required zones is required to be performed within 72 hours following an earthquake.

SNUBBERS (Continued)

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.8 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met.

FIRE SUPPRESSION SYSTEMS (Continued)

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-~~four hour~~ report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

The San Onofre Unit 2&3 fire pumps and water supplies, supply water to the San Onofre Unit 1 fire system.

3/4.7.9 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

ATTACHMENT B

INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

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

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.7.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

4.3.3.7.4 Following a seismic event (basemat acceleration greater than or equal to 0.05 g):

a. Within 2 hours each zone shown in Table 3.3-11 shall be inspected for fires, and

b. Within 72 hours an engineering evaluation shall be performed to verify the OPERABILITY of the fire detection system in each zone shown in Table 3.3-11.

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
1	Containment						
	Cable Tray Areas Elev 63'3"			10			
	Cable Tray Areas Elev 45'			9			
	Cable Tray Areas Elev 30'			4			
	Elevator Machinery Room			1			
	Combustible Oil Area						
	Two steam generator rooms				32		
	Charcoal Filter Area						
	Elev 45'	2					
2	Penetration						
	Elev 63'6"			12			
4	New Fuel Storage Area and Spent Fuel Pool Areas						
	Spent Fuel Pool		4				
	New Fuel Pool		3				
5	Control Building Elev 70'						
	Cable Riser Gallery Rm 423			2	24		
	Cable Riser Gallery Rm 449			3	24		
6	Control Building Elev 70'						
	Radiation Chemical Lab Rms 421, 420	1					
7	Radwaste Elev 63'6"						
	Chemical Storage Area Rm 503			1			
	Radwaste Control Panel Rm 513			1			
	Storage Area Rm 523			1			
	Hot Machine Shop	1					
8	Radwaste Elev 63'6"						
	Waste Decay Tank Rms 511A	None					
9	Fuel Handling Building Elev 45'						
	Emgy. A.C. Unit Rm 309-Train A	1		1			
	Emgy. A.C. Unit Rm 302-Train B	1		1			
10	Penetration						
	Elev 45'			6			

*The fire detection instruments located within the Containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
11	S.E.B. Roof and Main Steam Relief Valves	None		2 (Note 1)			
12	Control Building Elev 50'						
	Cable Riser Gallery Rm 305			3	42		
	Cable Riser Gallery Rm 315			3	40		
13A	Control Building Elev 50'						
	Emgy. HVAC Unit Rm 309A	1					
13B	Control Building Elev 50'						
	Emgy. HVAC Unit Rm 309B	1					
14	Radwaste Elev 24'						
	Boric Acid Makeup Tank Rm 204B	None					
	Boric Acid Makeup Tank Rm 204A	None					
15	Control Building Elev 50'						
	ESF Switchgear Rm 308A			2			
	ESF Switchgear Rm 308B			2			
16	Radwaste Elev 37' & 50' Ion Exchangers	None					
17	Diesel Generator Building						
	Train A			3		4	
	Train B			3		4	
18	Diesel Fuel Oil Storage Tank Underground Vaults	None					
20	Condensate Storage Tank T-121	None					
21	Nuclear Storage Tank T-104	None					
22	Auxiliary Feedwater Pump Room			2	9 (Note 2)	6	
23	Fuel Handling Bldg Elev 30' Spent Fuel Pools Heat Exchange Room 209	None					
28	Penetration Elev. 30'	2		8 (Note 1)			

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TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
29	Control Building Elev 30'						
	Cable Riser Gallery Rm 236			3	51		
	Cable Riser Gallery Rm 224			3	52		
30	Electrical Tunnel Elev 30'6"			13	50		
31	Control Building Elev 30'			29			
32A	Control Building Elev 30'						
	Fan Room Rm 219 & Corridor Rm 221	2		1			
32B	Control Building Elev 30'						
	Fan Room Rm 233 & Corridor Rm 234	2		1			
34	Radwaste Elev 9' & 24'						
	Secondary Radwaste Tank						
	Rms 126A,B & 127A,B	None					
35	Radwaste Elev 9' & 24'						
	Spent Resin Tank Rms 125A,B	None					
36	Fuel Handling Building Elev 17'6"						
	Spent Fuel Pool Pump Rm 107			2			
37	Radwaste Elev 24'						
	Letdown Heat Exchanger						
	Rms 209A,B	None					
38	Radwaste Elev 24'						
	Letdown Control Valve Rms 218A,B	None					
39	Radwaste Elev 24'						
	Filter Crvd Tank Rm 216	None					
40	Radwaste Elev 9' & 24'						
	Primary Radwaste Tank Rms 211A,D	None					
41	Control Building Elev 9'						
	Cable Spreading Rm 111A			17	36		
	Cable Spreading Rm 111B			14	36		
42	Control Building Elev 9'						
	Cable Riser Gallery Rm 110			6	44		
	Cable Riser Gallery Rm 112			6	39		

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TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
43	Control Building Elev 9'						
	Emgy. Chiller Rm 115			2			
	Emgy. Chiller Rm 117			2			
44	Intake Structure						
	Pump Rm T2-106			4			
	Pump Rm T3-106			4			
45	Penetration Area Elev 9' & 15'						
	Piping Penetration Area 15'			6 (Note 1)			
48	Safety Equipment Building 9' CCW HX and Piping Rm 022-025	None					
50	Radwaste Elev 9' Charging Pump Rms 106A-F			6			
51	Radwaste Elev 9' Boric Acid Makeup Tank Rms 105A-D	None					
53	Electrical Tunnel Elev 9'6", 11'6", (-) 2'6"			21	54		
54	Safety Eqpmt Bldg Elev 15'6" & 8' Shutdown HX Rms 003, 004, 016, 018	None					
55	Safety Eqpmt Bldg Elev 8' Chemical Storage Tank Rm 019			1			
56	Safety Eqpmt Bldg Elev 8' Component Cooling Water Surge Tank Rms 020, 021	None					
57	Safety Eqpmt Bldg Elev 15'6" Pump Rm 005			1			
58	Radwaste Elev 37' Reactor Trip System Rms 308A-D, 309-A-C			9			
59	Safety Eqpmt Bldg Elev 15'6" Pump Rm 001			1			

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TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
60	Safety Eqpmt Bldg Elev 15'6" Pump Rm 015			1			
61	Safety Eqpmt Bldg Elev 15'6" Component Cooling Water Pump Rms 006, 007, 008			3			
62	Radwaste Elev 50' Volume Control Valve Rooms			2 (Note 1)			
63	Control Building Elev 50' Corridor			12			
64	Control Building Elev 50' Vital Power Distribution Rms 310A-H			8			
65	Control Building Elev 50' Battery Rms 306B-J			8			
66	Control Building Elev 50' Evacuation Rm 311			1			
67	Radwaste Elev 63'6" Cable Riser Gallery Rm 506A Cable Riser Gallery Rm 506B			2 2	4 4		
68	Penetration 9' - 63'6" Cable Riser Shaft			1	21		
69	Safety Eqpmt Bldg Elev 5'3" Salt Water Cooling Piping Rm 010	None					
70	Radwaste Elev 24' Duct Shaft Rms 222A,B	None					
72	Control Building Elev 70' Corridor 401			4 (Note 3)			
75	Refueling Water Storage Tank T-005	None					
76	Refueling Water Storage Tank T-006	None					

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TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
78	Control Building Elev 9'						
	Corridor Rm 105			4			
79	Control Building Elev 50'						
	ESF Switchgear Rm 302A			2			
	ESF Switchgear Rm 302B			2			
80	Radwaste Elev 37' & 50'						
	Duct Shaft Rms	None					
81	Radwaste Elev 63'6"						
	Duct Shaft Rms 527A,B	None					
83	Salt Water Cooling Tunnel			6*			
84	Safety Eqpmt Bldg Elev 8'						
	HVAC Rm 017			3			
Technical Support Center (TSC)		5		1 (Note 3)			

*3 in UNIT 2, 3 in UNIT 3

Notes

1. On completion of DCP 3-403E
2. On completion of DCP 3-122M
3. On completion of DCP 2-403E

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PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.8.2 The spray and/or sprinkler systems listed in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas outside containment in which redundant systems or components could be damaged; for other areas outside containment, establish an hourly fire watch patrol.

b. With one or more of the above required spray and/or sprinkler systems inside containment inoperable, restore the system to OPERABLE status within 24 hours or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 7 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status

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

SURVEILLANCE REQUIREMENTS

4.7.8.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) outside of containment in the flow path is in its correct position.

b. At least once per 31 days during each COLD SHUTDOWN or REFUELING by verifying that each valve (manual, power-operated or automatic) inside containment in the flow path is in its correct position,

c. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

d. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a test signal, and

   b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. By a visual inspection of the dry pipe spray and wet pipe spray sprinkler headers to verify their integrity, and

SURVEILLANCE REQUIREMENTS (Continued)

3. By a visual inspection of each spray/sprinkler head to verify the spray pattern is not obstructed.

e. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

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TABLE 3.7-5

Safety Related Spray and/or Sprinkler Systems

Hazard	Location	No. of Systems	System Type
Reactor Coolant Pumps	Containment	4	Deluge-Water Spray
R.R. Tunnel	Fuel Hand. Bldg.	1	Wet Pipe
Truck Ramp	Radwaste Bldg.	1	Wet Pipe
Cable Tunnel	Section 1	1	Deluge-Water Spray
Cable Tunnel	Section 2	1	Deluge-Water Spray
Cable Tunnel	Section 3	1	Deluge-Water Spray
Cable Tunnel	Section 4	1	Deluge-Water Spray
Cable Tunnel	Section 5	1	Deluge-Water Spray
Cable Tunnel	Section 6	1	Deluge-Water Spray
Cable Tunnel	Section 7	1	Deluge-Water Spray
Cable Tunnel	Section 8	1	Deluge-Water Spray
Cable Tunnel	Section 9	1	Deluge-Water Spray
Cable Tunnel	Section 10	1	Deluge-Water Spray
Cable Tunnel Riser	Fuel Hand Bldg.	1	Deluge-Water Spray
Cable Gallery	Radwaste Bldg.	2*	Deluge-Water Spray
Cable Risers El. 9 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 30 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 50 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Risers El. 70 ft.	Control Bldg.	2*	Deluge-Water Spray
Cable Spreading Room	Control Bldg.	4*	Deluge-Water Spray
Emergency A.C. Unit - Train A	Fuel Handling Bldg.	1**	Deluge-Water Spray
Emergency A.C. Unit - Train B	Fuel Handling Bldg.	1**	Deluge-Water Spray
Diesel Generator	DG Bldg.	2	Pre-Action Sprinkler
HVAC Room 309A and 309D Corridors 301 and 303	Control Building 50'	1	Wet Pipe
Auxiliary Feedwater Pump	Tank Building 30'	1	Pre-action Sprinkler
		1#	Deluge-Water Spray
Fan Room 233 and Corridor 234	Control Building 30'	1	Wet Pipe
Salt Water Cooling Pumps	Intake Structure	1	Wet Pipe
Salt Water Cooling Tunnel			
CCW Heat Exchangers and Piping Room; A/C Room 017	Safety Equipment Bldg.	1	Wet Pipe
Corridor 401	Control Building 70'	1	Wet Pipe
Corridor 105	Control Building 9'	1	Wet Pipe

* One half of these Systems are designated Unit 3 but are required to be OPERABLE for Unit 2 operation.

** Charcoal filter deluge systems are manually actuated.

#On Completion of DCP 3-122M

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FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.8.3 The fire hose stations shown in Table 3.7-6 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-6 inoperable, route a fire hose to provide equivalent nozzle flow capacity to the unprotected area(s) from an OPERABLE hose station or alternate fire water supply, within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise provide the additional hose within 24 hours. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, a fire hose shall be stored in an area easily accessible to the unprotected area. Signs identifying the purpose and location of the fire hose and related valves shall be mounted above the hose and at the inoperable hose station.

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

SURVEILLANCE REQUIREMENTS

4.7.8.3 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.

   2. Removing the hose for inspection and re-racking, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

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TABLE 3.7-6

FIRE HOSE STATIONS

LOCATION	ELEVATION	STATION NUMBER
Containment Bldg. - Unit 3	63'-6"	131
Containment Bldg. - Unit 3	63'-6"	67
Containment Bldg. - Unit 3	63'-6"	74
Containment Bldg. - Unit 3	45'-0"	68
Containment Bldg. - Unit 3	45'-0"	70
Containment Bldg. - Unit 3	45'-0"	73
Containment Bldg. - Unit 3	30'-0"	64
Containment Bldg. - Unit 3	30'-0"	66
Containment Bldg. - Unit 3	30'-0"	72
Containment Bldg. - Unit 3	17'-6"	65
Containment Bldg. - Unit 3	17'-6"	69
Containment Bldg. - Unit 3	17'-6"	71
Electrical Penetration Area - Unit 3	45'-0"	124
Electrical Penetration Area - Unit 3	45'-0"	125
Electrical Penetration Area - Unit 3	63'-6"	126
Electrical Penetration Area - Unit 3	63'-6"	127
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	9'-0"	109
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	9'-0"	114
Cable Spreading Room-Auxiliary Bldg. Control Area	9'-0"	108
Cable Spreading Room-Auxiliary Bldg. Control Area	9'-0"	113
Cable Spreading Room Corridor-Auxiliary Bldg. Control Area	9'-0"	48
Cable Spreading Room Corridor-Auxiliary Bldg. Control Area	9'-0"	60
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	30'-0"	110
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	30'-0"	115
Corridor (North)-Auxiliary Bldg. Control Area	30'-0"	49
Corridor (South)-Auxiliary Bldg. Control Area	30'-0"	61
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	50'-0"	111
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	50'-0"	116
Corridor (North)-Auxiliary Bldg. Control Area	50'-0"	50
Corridor (South)-Auxiliary Bldg. Control Area	50'-0"	62
HVAC Room Corridor-Auxiliary Bldg. Control Area	50'-0"	56
HVAC Room Corridor-Auxiliary Bldg. Control Area	50'-0"	57
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	70'-0"	112
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	70'-0"	117
Fuel Handling Bldg.-Unit 3	63'-6"	128
Fuel Handling Bldg.-Unit 3	63'-6"	129

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PLANT SYSTEMS

3/4.7.9 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.9 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, ventilation duct, and piping penetration seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.9.1 Each of the above required fire doors shall be verified OPERABLE by:

a. Verifying at least once per 24 hours the position of each closed fire door and that doors with automatic hold-open and release mechanisms are free of obstructions.

b. Verifying at least once per 7 days the position of each locked closed fire door.

c. Performing a CHANNEL FUNCTIONAL TEST at least once per 31 days of the fire door supervision system.

d. Inspecting at least once per 6 months the automatic hold-open, release, and closing mechanism and latches.

e. Performing a functional test at least once per 18 months of the automatic hold-open, release, closing mechanisms and latches.

SURVEILLANCE REQUIREMENTS

4.7.9.2 At least once per 18 months the above required fire rated assemblies and penetration sealing devices other than fire doors shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/ and associated hardware.

c. Performing a visual inspection of at least 10% of each type (mechanical and electrical) of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

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room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

The OPERABILITY of the remote shutdown instrumentation in Panel L411 ensures that sufficient capability is available to permit shutdown and maintenance of COLD SHUTDOWN of the facility in the event of a fire in the cable spreading room, control room or remote shutdown panel, L042.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

Since the fire detectors are non-seismic, a plant visual inspection for fires is required within 2 hours following an earthquake ($\geq$ 0.02g). Since safe shutdown systems are protected by seismic Category I barriers rated at 2 and 3 hours, any fire after an earthquake should be detected by this inspection before safe shutdown systems would be affected. Additionally, to verify the continued OPERABILITY of fire detection systems after an earthquake, an engineering evaluation of the fire detection instrumentation in the required zones is required to be performed within 72 hours following an earthquake.

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SNUBBERS (Continued)

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.8 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met.

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FIRE SUPPRESSION SYSTEMS (Continued)

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a 24-hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

The San Onofre Unit 2 and 3 fire pumps and water supplies supply water to the San Onofre Unit 1 fire system.

3/4.7.9 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

ATTACHMENT C

INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11:

a. For areas outside containment accessible during plant operation, establish one of the following within 1 hour:

   1. The operability of the automatic suppression system in the area; or

   2. An hourly fire watch patrol for those areas without suppression.

b. For areas inside containment, inspect containment if accessible during during plant operation at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

c. The provisions of specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 a. The above required early warning detectors and the actuating flame detectors which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

b. The above required heat actuating detectors shall be tested as follows: At least one detector per signal string shall be tested every six months. The detector(s) selected for testing shall be rotated so that all detectors in a signal string are tested every five years.

c. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 1 week unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.7.3 Following a seismic event (basemat acceleration greater than or equal to 0.05g):

a. Within 2 hours each accessible zone shown in Table 3.3-11 located outside containment shall be inspected for fires, and

b. Within 1 hour monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5 until 4.3.3.7.3.c is completed, or within 8 hours inspect each accessible zone shown in Table 3.3-11 located inside containment, and

c. Within 72 hours perform an engineering evaluation to verify the OPERABILITY of the fire detection system in each zone shown in Table 3.3-11.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE***

Fire Area/Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
2-CO-15-1A*	Reactor Coolant Pump 002				8/8		
	Reactor Coolant Pump 004				8/8		
2-CO-15-1B*	Reactor Coolant Pump 001				8/8		
	Reactor Coolant Pump 003				8/8		
2-CO-15-1C*	Containment Area Quadrants 1,2,3 and 4 Elevation 30'-0" Elevation 45'-0"			 4/4 9/9			
	Charcoal Filter Elevation 45'-0"	1/1**					
2-CO-63-1D*	Operating Floor Elevation 63'-0"			10/10			
2-PE-9-2A	Penetration Bldg Elevation 9'-0"			3/4			
2-PE-(-18)-2B	Penetration Bldg Piping Area Elevation 18'-0"			2/2			
2-PE-30-2C	Penetration Bldg Piping Area Elevation 30'-0"			7/7			
	Charcoal Filter	1/1**					
2-PE-30-2D	Penetration Bldg Piping Area Elevation 30'-0"			2/2			

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
2-PE-45-3A	Penetration Bldg Eletrical Penetration Area Elevation 45'-0"			7/7			
2-PE-63-3B	Penetration Bldg Eletrical Penetration Area Elevation 63'-0"			10/12			
2-AC-9-5	Auxiliary Control Bldg Cable Spreading Room Elevation 9'-0"			13/17	27/36		
2-AC-9-14	Auxiliary Control Bldg Cable Riser Gallery Elevation 9'-0"			5/7	34/45		
2-AC-30-28	Auxiliary Control Bldg Cable Riser Gallery Elevation 30'-0"			3/3	40/53		
2-AC-50-35	Auxiliary Control Bldg Switchgear Room 2B Elevation 50'-0"			2/2			
2-AC-50-36	Auxiliary Control Bldg Cable Riser Gallery Elevation 50'-0"			1/1	10/13		
2-AC-50-37	Auxiliary Control Bldg Cable Riser Gallery Elevation 50'-0"			2/2	22/29		
2-AC-50-38	Auxiliary Control Bldg HVAC Room 2A Elevation 50'-0"	1/1					
2-AC-50-39	Auxiliary Control Bldg HVAC Room 2B Elevation 50'-0"	1/1					
2-AC-50-40	Auxiliary Control Bldg Switchgear Room 2A Elevation 50'-0"			2/2			

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
2-AC-50-44	Auxiliary Control Bldg Distribution Room 2B Elevation 50'-0"			1/1			
2-AC-50-45	Auxiliary Control Bldg Distribution Room 2D Elevation 50'-0"			1/1			
2-AC-50-46	Auxiliary Control Bldg Distribution Room 2C Elevation 50'-0"			1/1			
2-AC-50-47	Auxiliary Control Bldg Distribution Room 2A Elevation 50'-0"			1/1			
2-AC-50-48	Auxiliary Control Bldg Battery Room 2A Elevation 50'-0"			1/1			
2-AC-50-49	Auxiliary Control Bldg Battery Room 2C Elevation 50'-0"			1/1			
2-AC-50-50	Auxiliary Control Bldg Battery Room 2D Elevation 50'-0"			1/1			
2-AC-50-51	Auxiliary Control Bldg Battery Room 2B Elevation 50'-0"			1/1			
2-AC-70-63	Auxiliary Control Bldg Cable Riser Gallery Elevation 70'-0"			2/2	18/24		
2-AR-9-87	Auxiliary Radwaste Bldg Charging Pump Room Elevation 9'-0"			1/1			
2-AR-9-88	Auxiliary Radwaste Bldg Charging Pump Room Elevation 9'-0"			1/1			

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
2-AR-9-89	Auxiliary Radwaste Bldg Charging Pump Room Elevation 9'-0"			1/1			
2-AR-63-119	Auxiliary Radwaste Bldg Cable Riser Gallery Elevation 63'-6"			2/2	4/4		
2-SE-(-5)-135B	Safety Equipment Bldg Train B CCW Pump Room Elevation -5'-0"			1/1			
2-SE-(-5)-135C	Safety Equipment Bldg Spare CCW Pump Room Elevation -5'-0"			1/1			
2-SE-(-5)-135D	Safety Equipment Bldg Train C CCW Pump Room			1/1			
2-SE-(-15)-136	Safety Equipment Bldg A/C Room Elevation 8'-0"			3/3			
2-SE-(-15)-137A	Safety Equipment Bldg Safety Related Pump Room Elevation -15'-0"			1/1			
2-SE-(-15)-137B	Safety Equipment Bldg Safety Related Pump Room Elevation -15'-0"			1/1			
2-SE-(-15)-137C	Safety Equipment Bldg Safety Related Pump Room Elevation -15'-0"			1/1			
2-SE-8-140B	Safety Equipment Bldg Chemical Storage Room Elevation 8'-0"			1/1			
2-SE-30-142A	Safety Equipment Bldg Electrical Tunnel			17/17			
	Section 1				14/17		
	Section 2				3/4		

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
	Section 3				3/4		
	Section 4				3/4		
	Section 5				3/4		
	Section 6				6/8		
2-SE-30-145A	Safety Equipment Bldg Main Steam Relief Valves Elevation 30'-0"			2/2			
2-FH-17-122	Fuel Handling Bldg Fuel Pool Pump Room Elevation 17'-0"			2/2			
2-FH-17-123	Fuel Handling Bldg Spent Fuel Pool/Oper Floor Elevation 17'-0"		5/7				
2-FH-45-130	Fuel Handling Bldg A/C Room No. 2 Elevation 45'-0"			1/1			
	Charcoal Filter	1/1**					
2-FH-45-132	Fuel Handling Bldg A/C Room No. 1 Elevation 45'-0"			1/1			
	Charcoal Filter	1/1**					
2-CT-(-2)-142B	Electrical Cable Tunnel Elevation -2'-0"				16/21		
	Section 7				30/39		
	Section 8				7/9		
	Section 9				12/16		
	Section 10				8/10		

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
2-CT-16-142C	Cable Tunnel Cable Shaft Elevation 16'-0"			1/1	15/21		
2-DG-30-155	Diesel Generator Bldg Diesel Generator Room B Elevation 30'-0"			3/3		4/4	
2-DG-30-158	Diesel Generator Bldg Diesel Generator Room A Elevation 30'-0"			3/3		4/4	
2-TK-30-161A	Tank Building Auxiliary Feedwater Pump Room Elevation 30'-0"			2/2		5/6	
	AFW PUMPS P-504 & P-140				7/9		
COMMON AREAS							
2-AC-9-9	Auxiliary Control Bldg Emergency Chiller Room Elevation 9'-0"			1/2			
2-AC-9-11	Auxiliary Control Bldg Emergency Chiller Room Elevation 9'-0"			1/2			
2-AC-9-16	Auxiliary Control Bldg Corridor Elevation 9'-0"			6/7			
2-AC-30-20A	Auxiliary Control Bldg Control Room Elevation 30'-0"	8/8		11/11			
	Control Room Panels			19/19			
2-AC-30-20E	Auxiliary Control Bldg Lobby Elevation 30'-0"	1/1					
2-AC 30 23	Auxiliary Control Bldg Fan Room Elevation 30'-0"			1/1			

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
	Air Conditioner Charcoal Filter	1/1**					
	Emergency Ventilation Charcoal Filter	1/1**					
2-AC-30-26	Auxiliary Control Bldg Fan Room Elevation 30'-0"			1/1			
	Air Conditioner Charcoal Filter	1/1**					
	Emergency Ventilation Charcoal Filter	1/1**					
2-AC-50-29	Auxiliary Control Bldg Lobby/Monitor Control Room Elevation 50'-0"			11/12			
2-AC-50-43	Auxiliary Control Bldg Evacuation Room Elevation 50'-0"			1/1			
2-AC-70-64A	Auxiliary Control Bldg Corridor Elevation 70'-0"	3/3		3/4			
2-AC-70-64D	Auxiliary Control Bldg Office Area Elevation 70'-0"	2/2					
2-AC-70-64K	Auxiliary Control Bldg Rad. Chem. Lab Elevation 70'-0"	1/1					
2-AR-37-102A	Auxiliary Radwaste Bldg Corridor Elevation 37'-0"			9/9			
2-TB-(-9)-148E	Intake Structure Saltwater Cooling Tunnel Elevation -9'-0"			6/6			

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
2-TB-9-148F	Intake Structure Unit 2 Saltwater Cooling Pump Room Elevation 9'-0"			4/4			
3-TB-9-148F	Intake Structure Unit 3 Saltwater Cooling Pump Room Elevation 9'-0"			4/4			

* The fire detection instruments located within the Containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

** For charcoal filters, the thermistor strip detection system is required to be operable.

*** Minimum number of detectors required to be operable of the total number of detectors in an area or zone.

INSTRUMENTATION

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.8.2 The spray and/or sprinkler systems listed in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the required spray and/or sprinkler systems protecting redundant safe shutdown systems outside containment accessible during plant operation, inoperable, establish one of the following within 1 hour:

   1. a) The operability of the detection system in the area; and

      b) Backup fire suppression equipment if applicable;* or

   2. a) An hourly fire watch patrol for those areas without detection; and

      b) Backup fire suppression equipment if applicable.*

b. For other areas outside containment accessible during plant operation, establish the operability of the detection system in the area or an hourly fire watch patrol, within one 1 hour.

c. For areas inside containment with inoperable fire detection equipment and with one or more of the above required spray and/or sprinkler systems inoperable, inspect containment if accessible during plant operation at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

d. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

* Fire hose will be run within 4 hours of entering the ACTION statement if an operable water supply is not available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Fire hose will be supplied by the fire brigade responding to a fire if an operable water supply is available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Any additional backup fire suppression equipment is provided by the fire brigade responding to a fire.

SURVEILLANCE REQUIREMENTS

4.7.8.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) outside of containment in the flow path is in its correct position.

b. At least once per 31 days during each COLD SHUTDOWN exceeding 1 week or REFUELING by verifying that each valve (manual, power-operated or automatic) inside containment in the flow path is in its correct position.

c. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

d. At least once per 18 months:

   1. By performing a system functional test which includes simulated automatic actuation of the system except for the manually activated deluge-water spray system for the charcoal filters, and:

      a) Verifying that the automatic valves in the flow path actuate to their correct positions on a test signal, and

      b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

   2. By a visual inspection of the dry pipe spray and wet pipe spray sprinkler headers to verify their integrity, and

   3. By a visual inspection of each spray/sprinkler head to verify the spray pattern is not obstructed.

e. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

TABLE 3.7-5
REQUIRED SPRINKLER AND SPRAY SYSTEMS

Fire Area/Zone	Location of Protection	System Identifier	Type
2-CO-15-1A	Reactor Coolant Pump 002	8941	Deluge-Water Spray
	Reactor Coolant Pump 004	8944	Deluge-Water Spray
2-CO-15-1B	Reactor Coolant Pump 001	8942	Deluge-Water Spray
	Reactor Coolant Pump 003	8943	Deluge-Water Spray
2-CO-15-1C	Charcoal Filters in Recirc Filtration Unit Elevation 45'-0"	8945	Manually Activated Deluge-Water Spray
2-PE-30-2C	Charcoal Filters Elevation 30'-0"	8948	Manually Activated Deluge-Water Spray
2-AC-9-5	Cable Spreading Room Elevation 9'-0"	8981 8982	Deluge-Water Spray
2-AC-9-14	Cable Riser Gallery Elevation 9'-0"	5600	Deluge-Water Spray
2-AC-30-28	Cable Riser Gallery Elevation 30'-0"	5602	Deluge-Water Spray
2-AC-50-36	Cable Riser Gallery West Portion Elevation 50'-0"	5605	Deluge-Water Spray
2-AC-50-37	Cable Riser Gallery East Portion Elevation 50'-0"	5605	Deluge-Water Spray
2-AC-50-38	HVAC Room 2A Elevation 50'-0"	9007	Wet Pipe
2-AC-50-39	HVAC Room 2B Elevation 50'-0"	9007	Wet Pipe
2-AC-70-63	Cable Riser Gallery Elevation 70'-0"	5618	Deluge-Water Spray

TABLE 3.7-5 (Continued)
REQUIRED SPRINKLER AND SPRAY SYSTEMS

Fire Area/Zone	Location of Protection	System Identifier	Type
2-AR-63-119	Cable Riser Gallery Elevation 63'-6"	8971	Deluge-Water Spray
2-SE-(-5)-135A	Piping/Heat Exchanger Room Elevation 8'-0"	8997	Wet Pipe
2-SE-(-15)-136	A/C Room Elevation 8'-0"	8997	Wet Pipe
2-SE-30-142A	Electrical Tunnel Elevation 30'-0"		Deluge-Water Spray
	Section 1	8949	
	Section 2	8950	
	Section 3	8951	
	Section 4	8952	
	Section 5	8953	
	Section 6	8954	
2-FH-45-130	Charcoal Filters Emergency AC Unit 370 Elevation 45'-0"	8946	Manually Activated Deluge-Water Spray
2-FH-45-132	Charcoal Filters Emergency AC Unit 371 Elevation 45'-0"	8947	Manually Activated Deluge-Water Spray
2-CT-(-2)-142B	Electrical Cable Tunnel		Deluge-Water Spray
	Section 7	8955	
	Section 8	8956	
	Section 9	8957	
	Section 10	8936	

TABLE 3.7-5 (Continued)
REQUIRED SPRINKLER AND SPRAY SYSTEMS

Fire Area/Zone	Location of Protection	System Identifier	Type
2-CT-16-142C	Cable Shaft	8973	Deluge-Water Spray
2-DG-30-155	Diesel Generator Rm B Elevation 30'-0"	8975	Pre-Action Sprinkler
2-DG-30-158	Diesel Generator Rm A Elevation 30'-0"	8974	Pre-Action Sprinkler
2-TK-30-161A	Auxiliary Feedwater Pump Room Elevation 30'-0"	8987 8998	Pre-Action Sprinkler Deluge-Water Spray
COMMON			
2-AC-9-16	Corridor Elevation 9'-0"	9011	Wet Pipe
2-AC-30-23	Emergency AC Unit E-418	8985	Manually Activated Deluge-Water Spray
2-AC-30-26	Emergency AC Unit E-419	8979	Manually Activated Deluge-Water Spray
2-AC-30-27	Corridor Elevation 30'-0"	9009	Wet Pipe
2-AC-50-29	Lobby/Monitor Control Room	9007	Wet Pipe
2-AC-70-64A	Corridor Elevation 70'-0"	9005	Wet Pipe
2-TB-(-9)-148E	Saltwater Cooling Tunnel	9001	Wet Pipe
2-TB-9-148F	Unit 2 Saltwater Cooling Pump Room	9001	Wet Pipe
3-TB-9-148F	Unit 3 Saltwater Cooling Pump Room	9001	Wet Pipe

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.8.3 The fire hose stations shown in Table 3.7-6 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the above required fire hose stations inoperable, within 4 hours establish backup means of fire suppression if applicable* and within 24 hours post signs above the inoperable hose station(s) and related valves.

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

* Fire hose will be run within 4 hours of entering the ACTION statement if an operable water supply is not available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Fire hose will be supplied by the fire brigade responding to a fire if an operable water supply is available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Any additional backup fire suppression equipment is provided by the fire brigade responding to a fire.

SURVEILLANCE REQUIREMENTS

4.7.8.3 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.

   2. Removing the hose for inspection and re-racking, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

TABLE 3.7-6
FIRE HOSE STATIONS

Location	Elevation	Station Number
Unit 2 Containment	63'-6"	130
Unit 2 Containment	63'-6"	1
Unit 2 Containment	63'-6"	8
Unit 2 Containment	45'-0"	2
Unit 2 Containment	45'-0"	9
Unit 2 Containment	45'-0"	5
Unit 2 Containment	30'-0"	3
Unit 2 Containment	30'-0"	6
Unit 2 Containment	30'-0"	10
Unit 2 Containment	17'-6"	7
Unit 2 Containment	17'-6"	4
Unit 2 Containment	17'-6"	11
Unit 2 Electrical Penetration Area	63'-6"	122
Unit 2 Electrical Penetration Area	63'-6"	123
Unit 2 Electrical Penetration Area	45'-0"	120
Unit 2 Electrical Penetration Area	45'-0"	121
Cable Spreading Room (North) Auxiliary Control	9'-0"	108
Cable Riser Gallery (North) Auxiliary Control	9'-0"	109
Cable Riser Gallery (North) Auxiliary Control	30'-0"	110
Cable Riser Gallery (North) Auxiliary Control	50'-0"	111

TABLE 3.7-6 (Continued)
FIRE HOSE STATIONS

Location	Elevation	Station Number
Cable Riser Gallery (North) Auxiliary Control	70'-0"	112
Piping Room Safety Equipment	-15'-6"	31
Corridor Safety Equipment	-15'-6"	29
A/C Room Safety Equipment	8'-0"	28
Operating Floor Fuel Handling	63'-6"	118
Operating Floor Fuel Handling	63'-6"	119
Turbine Bldg	7'-0"	24
Intake Structure	9'-0"	95
Diesel Generator	30'-0"	97
Diesel Generator	30'-0"	98
COMMON		
Corridor Auxiliary Radwaste	9'-0"	32
Corridor Auxiliary Radwaste	9'-0"	36
Corridor Auxiliary Radwaste	9'-0"	40
Corridor Auxiliary Radwaste	9'-0"	44
Corridor Auxiliary Radwaste	24'-0"	33
Corridor Auxiliary Radwaste	24'-0"	37

TABLE 3.7-6 (Continued)
FIRE HOSE STATIONS

Location	Elevation	Station Number
Corridor Auxiliary Radwaste	24'-0"	45
Corridor Auxiliary Radwaste	37'-0"	34
Corridor Auxiliary Radwaste	37'-0"	38
Corridor Auxiliary Radwaste	37'-0"	41
Corridor Auxiliary Radwaste	37'-0"	46
Corridor Auxiliary Radwaste	50'-0"	102
Corridor Auxiliary Radwaste	50'-0"	106
Corridor Auxiliary Radwaste	63'-6"	103
Corridor Auxiliary Radwaste	63'-6"	107
Cable Spreading Rm Corridor Auxiliary Control	9'-0"	48
Cable Spreading Rm Corridor Auxiliary Control	9'-0"	60
Cable Spreading Rm Corridor Auxiliary Control	9'-0"	52
Corridor Auxiliary Control	30'-0"	49
Corridor Auxiliary Control	30'-0"	61

TABLE 3.7-6 (Continued)
FIRE HOSE STATIONS

Location	Elevation	Station Number
Lobby Auxiliary Control	30'-0"	53
Hall - Mezzanine Auxiliary Control	39'-2"	101
Lobby Auxiliary Control	50'-0"	50
Lobby - Motor Control Room Auxiliary Control	50'-0"	57
Lobby - Motor Control Room Auxiliary Control	50'-0"	56
Lobby Auxiliary Control	50'-0"	54
Lobby Auxiliary Control	50'-0"	62
Corridor Auxiliary Control	70'-0"	51
Corridor Auxiliary Control	70'-0"	55
Corridor Auxiliary Control	70'-0"	58
Corridor Auxiliary Control	70'-0"	59
Office Area Auxiliary Control	70'-0"	63
Roof (To Fan Room) Auxiliary Control	85'-0"	43

PLANT SYSTEMS

3/4.7.9 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.9 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, ventilation duct, and piping penetration seals) shall be OPERABLE.

APPLICABILITY: When equipment protected by the fire rated assemblies is required to be OPERABLE.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices separating portions of redundant systems important to safe shutdown within a fire area/zone accessible during plant operation, inoperable, within 1 hour either establish the OPERABILITY of the fire detectors or automatic suppression system in the area/zone, or establish a continuous fire watch.

b. With one or more of the above required fire rated assemblies and/or sealing devices separating other fire areas/zones accessible during plant operation inoperable, within 1 hour either establish the OPERABILITY of the fire detectors or automatic suppression system on at least one side of the inoperable assembly, or establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.9.1 Each of the above required fire doors accessible during plant operation shall be verified OPERABLE by:

a. Verifying at least once per 24 hours the position of each closed fire door and insure the doorways are free of obstructions.

b. Verifying at least once per 7 days the position of each locked closed fire door.

c. Inspecting at least once per 6 months the release and closing mechanism and latches.

4.7.9.2 At least once per 18 months the above required fire rated assemblies and penetration sealing devices other than fire doors shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/and associated hardware.

c. Performing a visual inspection of at least 10% of each type (mechanical and electrical) of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

The OPERABILITY of the remote shutdown instrumentation in Panel L411 ensures that sufficient capability is available to permit shutdown and maintenance of COLD SHUTDOWN of the facility in the event of a fire in the cable spreading room, control room or remote shutdown panel, L042.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident, "December 1975 and NUREG 0578, "TMI-2 Lessions Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols or establishment of the automatic suppression system in the affected areas is required to provide detection or suppression capability until the inoperable instrumentation is restored to OPERABILITY. Areas accessible during plant operation are defined as areas that do not pose radiation and/or other life-threatening safety hazards.

Since the fire detectors are non-seismic, a plant visual inspection for fires is required within 2 hours following an earthquake ($\geq$ 0.05g). Since safe shutdown systems are protected by Seismic Category I barriers rated at 2 and 3 hours, any fire after an earthquake should be detected by this inspection before safe shutdown systems would be affected. Additionally, to verify the continued OPERABILITY of fire detection systems after an earthquake, an engineering evaluation of the fire detection instrumentation in the required zones is required to be performed within 72 hours following an earthquake.

SNUBBERS (Continued)

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requiremnts for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from by-product, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.8 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression sytems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate fire protection features are required until the operable equipment is restored to service. In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

FIRE SUPPRESSION SYSTEMS (Continued)

The surveillances requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. Areas accessible during plant operation are defined as areas that do not pose radiation and/or other life-threatening safety hazards.

The San Onofre Unit 2 and 3 fire pumps and water supplies supply water to the San Onofre Unit 1 fire system.

3/4.7.9 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

Areas accessible during plant operation are defined as areas that do not pose radiation and/or other life-threatening safety hazards.

ATTACHMENT D

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11:

a. For areas outside containment accessible during plant operation, establish one of the following within 1 hour:

   1. The operability of the automatic suppression system in the area; or

   2. An hourly fire watch patrol for those areas without suppression.

b. For areas inside containment, inspect containment if accessible during during plant operation at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

c. The provisions of specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 a. The above required early warning detectors and the actuating flame detectors which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

b. The above required heat actuating detectors shall be tested as follows: At least one detector per signal string shall be tested every six months. The detector(s) selected for testing shall be rotated so that all detectors in a signal string are tested every five years.

c. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 1 week unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.7.3 Following a seismic event (basemat acceleration greater than or equal to 0.05g):

a. Within 2 hours each accessible zone shown in Table 3.3-11 located outside containment shall be inspected for fires, and

b. Within 1 hour monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5 until 4.3.3.7.3.c is completed, or within 8 hours inspect each accessible zone shown in Table 3.3-11 located inside containment, and

c. Within 72 hours perform an engineering evaluation to verify the OPERABILITY of the fire detection system in each zone shown in Table 3.3-11.

TABLE 3.3-11
FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE***

Fire Area/Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
3-CO-15-1A*	Reactor Coolant Pump 002				9/9		
	Reactor Coolant Pump 004				8/8		
3-CO-15-1B*	Reactor Coolant Pump 001				8/8		
	Reactor Coolant Pump 003				8/8		
3-CO-15-1C*	Containment Area Quadrants 1,2,3 and 4 Elevation 30'-0" Elevation 45'-0"			 4/4 9/9			
	Charcoal Filters Elevation 45'-0"	1/1**					
3-CO-63-1D*	Containment Area Operating Floor Elevation 63'-0"			10/10			
3-PE-9-2A	Penetration Bldg Elevation 9'0"			3/4			
3-PE-(-18)-2B	Penetration Bldg Piping Area Elevation -18'-0"			2/2			
3-PE-30-2C	Penetration Bldg Piping Area Elevation 30'-0"			7/7			
	Charcoal Filter	1/1**					
3-PE-30-2D	Penetration Bldg Piping Area Elevation 30'-0"			2/2			

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
3-PE-45-3A	Penetration Bldg Electrical Penetration Area Elevation 45'-0"			7/7			
3-PE-63-3B	Penetration Bldg Electrical Penetration Area Elevation 63'-0"			10/12			
3-AC-9-6	Auxiliary Control Bldg Cable Spreading Room Elevation 9'-0"			11/14	27/36		
3-AC-9-7	Auxiliary Control Bldg Cable Riser Gallery Elevation 9'-0"			5/7	30/39		
3-AC-30-21	Auxiliary Control Bldg Cable Riser Gallery Elevation 30'-0"			3/3	39/52		
3-AC-50-30	Auxiliary Control Bldg HVAC Room 3B Elevation 50'-0"	1/1					
3-AC-50-31	Auxiliary Control Bldg HVAC Room 3A Elevation 50'-0"	1/1					
3-AC-50-32	Auxiliary Control Bldg Cable Riser Gallery Elevation 50'-0"			2/2	21/27		
3-AC-50-33	Auxiliary Control Bldg Cable Riser Gallery Elevation 50'-0"			1/1	10/13		
3-AC-50-34	Auxiliary Control Bldg Switchgear Room 3B Elevation 50'-0"			2/2			
3-AC-50-52	Auxiliary Control Bldg Battery Room 3B Elevation 50'-0"			1/1			
3-AC-50-53	Auxiliary Control Bldg Battery Room 3D Elevation 50'-0"			1/1			

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
3-AC-50-54	Auxiliary Control Bldg Battery Room 3C Elevation 50'-0"			1/1			
3-AC-50-55	Auxiliary Control Bldg Battery Room 3A Elevation 50'-0"			1/1			
3-AC-50-56	Auxiliary Control Bldg Distribution Room 3A Elevation 50'-0"			1/1			
3-AC-50-57	Auxiliary Control Bldg Distribution Room 3C Elevation 50'-0"			1/1			
3-AC-50-58	Auxiliary Control Bldg Distribution Room 3D Elevation 50'-0"			1/1			
3-AC-50-59	Auxiliary Control Bldg Distribution Room 3B Elevation 50'-0"			1/1			
3-AC-50-60	Auxiliary Control Bldg Switchgear Room 3A Elevation 50'-0"			2/2			
3-AC-70-65	Auxiliary Control Bldg Cable Riser Gallery Elevation 70'-0"			3/3	18/24		
3-AR-9-91	Auxiliary Radwaste Bldg Charging Pump Room Elevation 9'-0"			1/1			
3-AR-9-92	Auxiliary Radwaste Bldg Charging Pump Room Elevation 9'-0"			1/1			
3-AR-9-93	Auxiliary Radwaste Bldg Charging Pump Room Elevation 9'-0"			1/1			

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
3-AR-63-118	Auxiliary Radwaste Bldg Cable Tray Gallery Elevation 63'-0"			2/2	4/4		
3-SE-(-5)-135B	Safety Equipment Bldg Train B CCW Pump Room Elevation -5'-0"			1/1			
3-SE-(-5)-135C	Safety Equipment Bldg Spare CCW Pump Room Elevation -5'-0"			1/1			
3-SE-(-5)-135D	Safety Equipment Bldg Train A CCW Pump Room Elevation -5'-0"			1/1			
3-SE-(-15)-136	Safety Equipment Bldg A/C Room Elevation 8'-0"			3/3			
3-SE-(-15)-137A	Safety Equipment Bldg Safety Related Pump Room Elevation -15'-0"			1/1			
3-SE-(-15)-137B	Safety Equipment Bldg Safety Related Pump Room Elevation -15'-0"			1/1			
3-SE-(-15)-137C	Safety Equipment Bldg Safety Related Pump Room Elevation -15'-0"			1/1			
3-SE-8-140B	Safety Equipment Bldg Chemical Storage Room Elevation 8'-0"			1/1			
3-SE-30-142A	Safety Equipment Bldg Electrical Tunnel			17/17			
	Section 1				12/15		
	Section 2				3/4		
	Section 3				3/4		

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
	Section 4				3/4		
	Section 5				3/4		
	Section 6				6/8		
3-SE-30-145A	Safety Equipment Bldg Main Steam Relief Valves Elevation 30'-0"			2/2			
3-FH-17-122	Fuel Handling Bldg Fuel Pool Pump Room Elevation 17'-0"			2/2			
3-FH-17-123	Fuel Handling Bldg Spent Fuel Pool/Oper Floor Elevation 17'-0"		5/7				
3-FH-45-130	Fuel Handling Bldg A/C Room No. 2 Elevation 45'-0"			1/1			
	Charcoal Filter	1/1**					
3-FH-45-132	Fuel Handling Bldg A/C Room No. 1 Elevation 45'-0"			1/1			
	Charcoal Filter	1/1**					
3-CT-(-2)-142B	Electrical Cable Tunnel Elevation -2'-0"			16/21			
	Section 7				29/38		
	Section 8				7/9		
	Section 9				12/16		
	Section 10				8/10		
3-CT-16-142C	Cable Tunnel Cable Shaft Elevation 16'-0"			1/1	15/21		

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
3-DG-30-155	Diesel Generator Bldg Diesel Generator Room B Elevation 30'-0"			3/3		4/4	
3-DG-30-158	Diesel Generator Bldg Diesel Generator Room A Elevation 30'-0"			3/3		4/4	
3-TK-30-161A	Tank Bldg Auxiliary Feedwater Pump Room Elevation 30'-0"			2/2		5/6	
	AFW PUMPS P-504 & P-140				7/9		
<u>COMMON</u>							
2-AC-9-9	Auxiliary Control Bldg Emergency Chiller Room Elevation 9'-0"			1/2			
2-AC-9-11	Auxiliary Control Bldg Emergency Chiller Room Elevation 9'-0"			1/2			
2-AC-9-16	Auxiliary Control Bldg Corridor Elevation 9'-0"			6/7			
2-AC-30-20A	Auxiliary Control Bldg Control Room Elevation 30'-0"	8/8		11/11			
	Control Room Panels			19/19			
2-AC-30-20E	Auxiliary Control Bldg Lobby Elevation 30'-0"	1/1					
2-AC-30-23	Auxiliary Control Bldg Fan Room Elevation 30'-0"			1/1			
	Air Conditioner Charcoal Filter	1/1**					

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
	Emergency Ventilation Charcoal Filter	1/1**					
2-AC-30-26	Auxiliary Control Bldg Fan Room Elevation 30'-0"			1/1			
	Air Conditioner Charcoal Filter	1/1**					
	Emergency Ventilation Charcoal Filter	1/1**					
2-AC-50-29	Auxiliary Control Bldg Lobby/Monitor Control Room Elevation 50'-0"			11/12			
2-AC-50-43	Auxiliary Control Bldg Evacuation Room Elevation 50'-0"			1/1			
2-AC-70-64A	Auxiliary Control Bldg Corridor Elevation 70'-0"	3/3		3/4			
2-AC-70-64D	Auxiliary Control Bldg Office Area Elevation 70'-0"	2/2					
2-AC-70-64K	Auxiliary Control Bldg Rad. Chem. Lab Elevation 70'-0"	1/1					
2-AR-37-102A	Auxiliary Radwaste Bldg Corridor Elevation 37'-0"			9/9			
2-TB-(-9)-148E	Intake Structure Saltwater Cooling Tunnel Elevation -9'-0"			6/6			
2-TB-9-148F	Intake Structure Unit 2 Saltwater Cooling Pump Room Elevation 9'-0"			4/4			

TABLE 3.3-11 (Continued)

Fire Area/Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
3-TB-9-148F	Intake Structure Unit 3 Saltwater Cooling Pump Room Elevation 9'-0"			4/4			

* The fire detection instruments located within the Containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

** For charcoal filters, the thermistor strip detection system is required to be operable.

*** Minimum number of detectors required to be operable of the total number of detectors in an area or zone.

INSTRUMENTATION

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.8.2 The spray and/or sprinkler systems listed in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the required spray and/or sprinkler systems protecting redundant safe shutdown systems outside containment accessible during plant operation, inoperable, establish one of the following within 1 hour:

   1. a) The operability of the detection system in the area; and

      b) Backup fire suppression equipment if applicable;* or

   2. a) An hourly fire watch patrol for those areas without detection; and

      b) Backup fire suppression equipment if applicable.*

b. For other areas outside containment accessible during plant operation, establish the operability of the detection system in the area or an hourly fire watch patrol, within one 1 hour.

c. For areas inside containment with inoperable fire detection equipment and with one or more of the above required spray and/or sprinkler systems inoperable, inspect containment if accessible during plant operation at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

d. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

* Fire hose will be run within 4 hours of entering the ACTION statement if an operable water supply is not available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Fire hose will be supplied by the fire brigade responding to a fire if an operable water supply is available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Any additional backup fire suppression equipment is provided by the fire brigade responding to a fire.

SURVEILLANCE REQUIREMENTS

4.7.8.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) outside of containment in the flow path is in its correct position.

b. At least once per 31 days during each COLD SHUTDOWN exceeding 1 week or REFUELING by verifying that each valve (manual, power-operated or automatic) inside containment in the flow path is in its correct position.

c. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

d. At least once per 18 months:

   1. By performing a system functional test which includes simulated automatic actuation of the system except for the manually activated deluge-water spray system for the charcoal filters, and:

      a) Verifying that the automatic valves in the flow path actuate to their correct positions on a test signal, and

      b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

   2. By a visual inspection of the dry pipe spray and wet pipe spray sprinkler headers to verify their integrity, and

   3. By a visual inspection of each spray/sprinkler head to verify the spray pattern is not obstructed.

e. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

TABLE 3.7-5
REQUIRED SPRINKLER AND SPRAY SYSTEMS

Fire Area/Zone	Location of Protection	System Identifier	Type
3-CO-15-1A	Reactor Coolant Pump 002	8941	Deluge-Water Spray
	Reactor Coolant Pump 004	8944	Deluge-Water Spray
3-CO-15-1B	Reactor Coolant Pump 001	8942	Deluge-Water Spray
	Reactor Coolant Pump 003	8943	Deluge-Water Spray
3-CO-15-1C	Charcoal Filters in Recirc Filtration Unit Elevation 45'-0"	8945	Manually Activated Deluge-Water Spray
3-PE-30-2C	Charcoal Filters in Elevation 45'-0"	8948	Manually Activated Deluge-Water Spray
3-AC-9-6	Cable Spreading Room Elevation 9'-0"	8983 8984	Deluge-Water Spray
3-AC-9-7	Cable Riser Gallery Elevation 9'-0"	5621	Deluge-Water Spray
3-AC-30-21	Cable Riser Gallery Elevation 30'-0"	5632	Deluge-Water Spray
3-AC-50-30	HVAC Room 3B Elevation 50'-0"	9007	Wet Pipe
3-AC-50-31	HVAC Room 3A Elevation 50'-0"	9007	Wet Pipe
3-AC-50-32	Cable Riser Gallery East Portion Elevation 50' - 0"	5639	Deluge-Water Spray
3-AC-50-33	Cable Riser Gallery West Portion Elevation 50' - 0"	5639	Deluge-Water Spray
3-AC-70-65	Cable Riser Gallery Elevation 70' - 0"	5682	Deluge-Water Spray
3-AR-63-118	Cable Riser Gallery Elevation 63' - 0"	8971	Deluge-Water Spray

TABLE 3.7-5 (Continued)
REQUIRED SPRINKLER AND SPRAY SYSTEMS

Fire Area/Zone	Location of Protection	System Identifier	Type
3-SE-(-5)-135A	Piping/Heat Exchanger Room Elevation 8' - 0"	8997	Wet Pipe
3-SE-(-15)-136	AC Room Elevation 8' - 0"	8997	Wet Pipe
3-SE-30-142A	Electrical Tunnel Elevation 30'-0"		Deluge-Water Spray
	Section 1	8949	
	Section 2	8950	
	Section 3	8951	
	Section 4	8952	
	Section 5	8953	
	Section 6	8954	
3-FH-45-130	Charcoal Filter Emergency AC Unit E-370 Elevation 45'-0"	8946	Manually Activated Deluge-Water Spray
3-FH-45-132	Charcoal Filter Emergency AC Unit E-371 Elevation 45'-0"	8947	Manually Activated Deluge-Water Spray
3-CT-(-2)-142B	Electrical Cable Tunnel		Deluge-Water Spray
	Section 7	8955	
	Section 8	8956	
	Section 9	8957	
	Section 10	8936	
3-CT-16-142C	Cable Shaft	8973	Deluge-Water Spray

TABLE 3.7-5 (Continued)
REQUIRED SPRINKLER AND SPRAY SYSTEMS

Fire Area/Zone	Location of Protection	System Identifier	Type
3-DG-30-155	Diesel Generator Rm. B Elevation 30'-0"	8975	Pre-Action Sprinkler
3-DG-30-158	Diesel Generator Rm. A Elevation 30'-0"	8974	Pre-Action Sprinkler
3-TK-30-161A	Auxiliary Feedwater	8987	Pre-Action Sprinkler
	Pump Room Elevation 30'-0"	8998	Deluge-Water Spray
COMMON			
2-AC-9-16	Corridor Elevation 9' - 0"	9011	Wet Pipe
2-AC-30-23	Emergency AC Unit E-418	8985	Manually Activated Deluge-Water Spray
2-AC-30-26	Emergency AC Unit E-419	8979	Manually Activated Deluge-Water Spray
2-AC-30-27	Corridor Elevation 30' - 0"	9009	Wet Pipe
2-AC-50-29	Lobby/Monitor Control Room	9007	Wet Pipe
2-AC-70-64A	Corridor Elevation 70' - 0"	9005	Wet Pipe
2-TB-(-9)-148E	Saltwater Cooling Tunnel	9001	Wet Pipe
2-TB-9-148F	Unit 2 Saltwater Cooling Pump Room	9001	Wet Pipe
3-TB-9-148F	Unit 3 Saltwater Cooling Pump Room	9001	Wet Pipe

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.8.3 The fire hose stations shown in Table 3.7-6 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the above required fire hose stations inoperable, within 4 hours establish backup means of fire suppression if applicable* and within 24 hours post signs above the inoperable hose station(s) and related valves.

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

* Fire hose will be run within 4 hours of entering the ACTION statement if an operable water supply is not available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Fire hose will be supplied by the fire brigade responding to a fire if an operable water supply is available within 300 feet of the area containing the inoperable spray and/or sprinkler system. Any additional backup fire suppression equipment is provided by the fire brigade responding to a fire.

SURVEILLANCE REQUIREMENTS

4.7.8.3 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.

2. Removing the hose for inspection and re-racking, and

3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

TABLE 3.7-6
FIRE HOSE STATIONS

Location	Elevation	Station Number
Unit 3 Containment	63'-6"	131
Unit 3 Containment	63'-6"	67
Unit 3 Containment	63'-6"	74
Unit 3 Containment	45'-0"	68
Unit 3 Containment	45'-0"	70
Unit 3 Containment	45'-0"	73
Unit 3 Containment	30'-0"	66
Unit 3 Containment	30'-0"	64
Unit 3 Containment	30'-0"	72
Unit 3 Containment	17'-6"	65
Unit 3 Containment	17'-6"	69
Unit 3 Containment	17'-6"	71
Unit 3 Electrical Penetration Area	45'-0"	124
Unit 3 Electrical Penetration Area	45'-0"	125
Unit 3 Electrical Penetration Area	63'-6"	126
Unit 3 Electrical Penetration Area	63'-6"	127
Piping Room Safety Equipment	-15'-6"	94
Corridor Safety Equipment	-15'-6"	92
A/C Room Safety Equipment	8'-0"	91
Operating Floor Fuel Handling	63'-6"	128

TABLE 3.7-6 (Continued)
FIRE HOSE STATIONS

Location	Elevation	Station Number
Operating Floor Fuel Handling	63'-6"	129
Turbine Bldg	7'-0"	87
Intake Structure	9'-0"	96
Cable Riser Gallery (South) Auxiliary Control	9'-0"	113
Cable Riser Gallery (South) Auxiliary Control	9'-0"	114
Cable Riser Gallery (South) Auxiliary Control	30'-0"	115
Cable Riser Gallery (South) Auxiliary Control	50'-0"	116
Cable Riser Gallery (South) Auxiliary Control	70'-0"	117
Diesel Generator	30'-0"	99
Diesel Generator	30'-0"	100
COMMON		
Corridor Auxiliary Radwaste	9'-0"	32
Corridor Auxiliary Radwaste	9'-0"	36
Corridor Auxiliary Radwaste	9'-0"	40
Corridor Auxiliary Radwaste	9'-0"	44
Corridor Auxiliary Radwaste	24'-0"	33
Corridor Auxiliary Radwaste	24'-0"	37

TABLE 3.7-6 (Continued)
FIRE HOSE STATIONS

Location	Elevation	Station Number
Corridor Auxiliary Radwaste	24'-0"	45
Corridor Auxiliary Radwaste	37'-0"	34
Corridor Auxiliary Radwaste	37'-0"	38
Corridor Auxiliary Radwaste	37'-0"	41
Corridor Auxiliary Radwaste	37'-0"	46
Corridor Auxiliary Radwaste	50'-0"	102
Corridor Auxiliary Radwaste	50'-0"	106
Corridor Auxiliary Radwaste	63'-6"	103
Corridor Auxiliary Radwaste	63'-6"	107
Cable Spreading Rm Corridor Auxiliary Control	9'-0"	48
Cable Spreading Rm Corridor Auxiliary Control	9'-0"	60
Cable Spreading Rm Corridor Auxiliary Control	9'-0"	52
Corridor Auxiliary Control	30'-0"	49
Corridor Auxiliary Control	30'-0"	61
Lobby Auxiliary Control	30'-0"	53

TABLE 3.7-6 (Continued)
FIRE HOSE STATIONS

Location	Elevation	Station Number
Hall - Mezzanine Auxiliary Control	39'-2"	101
Lobby Auxiliary Control	50'-0"	50
Lobby - Motor Control Room Auxiliary Control	50'-0"	57
Lobby - Motor Control Room Auxiliary Control	50'-0"	56
Lobby Auxiliary Control	50'-0"	54
Lobby Auxiliary Control	50'-0"	62
Corridor Auxiliary Control	70'-0"	51
Corridor Auxiliary Control	70'-0"	55
Corridor Auxiliary Control	70'-0"	58
Corridor Auxiliary Control	70'-0"	59
Office Area Auxiliary Control	70'-0"	63
Roof (To Fan Room) Auxiliary Control	85'-0"	43

PLANT SYSTEMS

3/4.7.9 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.9 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, ventilation duct, and piping penetration seals) shall be OPERABLE.

APPLICABILITY: When equipment protected by the fire rated assemblies is required to be OPERABLE.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices separating portions of redundant systems important to safe shutdown within a fire area/zone accessible during plant operation, inoperable, within 1 hour either establish the OPERABILITY of the fire detectors or automatic suppression system in the area/zone, or establish a continuous fire watch.

b. With one or more of the above required fire rated assemblies and/or sealing devices separating other fire areas/zones accessible during plant operation inoperable, within 1 hour either establish the OPERABILITY of the fire detectors or automatic suppression system on at least one side of the inoperable assembly, or establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.9.1 Each of the above required fire doors accessible during plant operation shall be verified OPERABLE by:

a. Verifying at least once per 24 hours the position of each closed fire door and insure the doorways are free of obstructions.

b. Verifying at least once per 7 days the position of each locked closed fire door.

c. Inspecting at least once per 6 months the release and closing mechanism and latches.

4.7.9.2 At least once per 18 months the above required fire rated assemblies and penetration sealing devices other than fire doors shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/and associated hardware.

c. Performing a visual inspection of at least 10% of each type (mechanical and electrical) of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

The OPERABILITY of the remote shutdown instrumentation in Panel L411 ensures that sufficient capability is available to permit shutdown and maintenance of COLD SHUTDOWN of the facility in the event of a fire in the cable spreading room, control room or remote shutdown panel, L042.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident, "December 1975 and NUREG 0578, "TMI-2 Lessions Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols or establishment of the automatic suppression system in the affected areas is required to provide detection or suppression capability until the inoperable instrumentation is restored to OPERABILITY. Areas accessible during plant operation are defined as areas that do not pose radiation and/or other life-threatening safety hazards.

Since the fire detectors are non-seismic, a plant visual inspection for fires is required within 2 hours following an earthquake ($\geq$ 0.05g). Since safe shutdown systems are protected by Seismic Category I barriers rated at 2 and 3 hours, any fire after an earthquake should be detected by this inspection before safe shutdown systems would be affected. Additionally, to verify the continued OPERABILITY of fire detection systems after an earthquake, an engineering evaluation of the fire detection instrumentation in the required zones is required to be performed within 72 hours following an earthquake.

SNUBBERS (Continued)

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requiremnts for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from by-product, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.8 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression sytems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate fire protection features are required until the operable equipment is restored to service. In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

FIRE SUPPRESSION SYSTEMS (Continued)

The surveillances requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. Areas accessible during plant operation are defined as areas that do not pose radiation and/or other life-threatening safety hazards.

The San Onofre Unit 2 and 3 fire pumps and water supplies supply water to the San Onofre Unit 1 fire system.

3/4.7.9 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

Areas accessible during plant operation are defined as areas that do not pose radiation and/or other life-threatening safety hazards.