KGE THE ELECTRIC COMPANY

KANSAS GAS AND ELECTRIC COMPANY

GLENN L. KOESTER
VICE PRESIDENT - NUCLEAR

December 3, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

KMLNRC 84-211
Re: Docket No. STN 50-482
Ref: Letter dated 11/07/84 from DGEisenhut, NRC,
to GLKoester, KG&E
Subj: Technical Specifications

Dear Mr. Denton:

The Reference provided the "final draft" of the Wolf Creek Technical Specifications for KG&E review and affirmation.

KG&E's review of the Technical Specifications is not yet complete. However, to aid the NRC provided herewith are interim comments on the draft. KG&E's review will be complete with the results forwarded to the NRC on or before December 10, 1984.

Yours very truly,

Glenn L Koester

GLK:bb
Attach
xc:PO'Connor (2)
HBundy

B001
1/40

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TECHNICAL SPECIFICATION CHANGES

Page	Section	Reason for Change
XX11	NA	Re orting requirements was completed on previous p je and not continued on this page.
2-4	Table 2.2-1	Cnange not made with other changes on this page.
3/4 1-10	Footnote	Change required to achieve correct meaning for footnote.
3/4 1-23	NA	To be consistent with balance of Tech Spec (All other references to three-loop operation have been deleted.)
3/4 2-10	4.2.3.6	Specification not required since WCGS will utilize the leading edge flow meter to determine degradation of the feedwater venturi.
3/4 2-11	Action a.4	To be consistent with Tech Spec format.
3/4 3-2&5	Table 3.3-1	Justification with change in attachment.
3/4 3-6	Action 5	Change time frame to be consistent with Spec. 3.1.1.2. Valve lineups are controlled by locked valve lists and surveillance not necessary.
3/4 3-13	Action b.2	Typo.
3/4 3-27 & 3-28	Table 3.3-4 8.a & b. and Table notation	Reevaluation of design criteria. Time delay changes were made to be compatiable with Table 4.3-2.
3/4 3-30	4.a.6	Consistency with rest of table.
3/4 3-35	Table 4.3-2	Consistency with rest of table.
3/4 3-40	Table 3.3-6 2.b	Correction not picked up. To correctly identify instrumentation.
3/4 3-54	Table 3.3-10 Item 18	To correctly identify instrument.
3/4 3-55	Table 4.3.7 ***Footnote	Typo.
3/4 3-58, 59,60,& 61	Table 3.3-11	Provide additional information.
3/4 3-71	Action 42 Action 44	Typo. Typo.

TECHNICAL SPECIFICATION CHANGES (Continued)

Page	Section	Reason for Change
3/4 3-74	Table 4.3-9 Notations (1)d. & (2)	Typos.
3/4 4-2	**Footnote	Correctly identify Specification.
3/4 4-10	3.4.4 Action a., b., & c.	Justification with change in attachment.
3/4 4-34	Action a.	Consistent with wording in Callaway Tech Spec.
3/4 6-2	NA	Info to be provided after Performance of Test.
3/4 6-11	3.6.1.7 Action b.	Justification provided with change in attachment.
3/4 6-21	Table 3.6-1 1. P-69	Valve functions misidentified.
3/4 6-25	Table 3.6-1 6. P-76	Correctly identify function.
	*Footnote	Typo.
3/4 6-27 & 28	Table 3.6-1	Incorrect Nomenclature.
3/4 6-29 & 30	Table 3.6-1 9.	These valves are not considered as containment isolation valves in the FSAR. See justification provided with change in attachment.
3/4 7-28	4.7.10.1.1f.	a. System does not have automatic valves in flow path b. Per FSAR, 80 psig is correct value. c. Change required to clarify requirement and to correspond with design.
3/4 7-34	Table 3.7-3	Consistency with rest of table.
3/4 7-35	Table 3.7-3	Provide necessary information concerning equipment.
3/4 8-3, 4, & 5	4.8.1.1.2a.4), f.2), f.4) b), f.5), f.6) b), f.7).	Changed to reflect design and for compatibility with Specifications 3.8.3.1 and 3.8.3.2.

TECHNICAL SPECIFICATION CHANGES (Continued)

Page	Section	Reason for Change
3/4 8-8	Action Statement	To be consistent with rest of Tech. Spec.
3/4 8-9	3.8.2.1a.	Both are required, not one "or" the other.
3/4 8-11	Table 4.8-2	Clarification.
3/4 8-39	Table 3.8-1	Typos.
3/4 8-40	Table 3.8-1	Typo.
3/4 9-18	4.9.13b.2)	Typo.
3/4 10-4	4.10.4.3	Typo.
3/4 11-3	Table 4.11-1 Notation (2)	Justification with change in attachment.
3/4 11-9 3/4 11-11	Table 4.11-2	Justification provided with change in attachment.
3/4 11-16	3.11.2.6	Justification provided with change in attachment.
3/4 11-18	3/4.11.4	This page missing from Final Draft.
3/4 12-1 & 2	3.12.1 Action C	Justification provided with change in attachment.
3/4 12-4	Table 3.12-1, 3.a.	Justification provided with change in attachment.
3/4 12-8	Table 3.12-1 Notation (7)	Justification provided with change in attachment.
3/4 12-14	3/4.12.3	This page missing from Final Draft.
B 3/4 2-4	3/4.2.2 & 3/4.2.3	Clarify reference.
B 3/4 2-5	3/4.2.2 and 3/4.2.3	Justification with change in attachment.

TECHNICAL SPECIFICATION CHANGES (Continued)

Page	Section	Reason for Change
B 3/4 3-4	3/4.3.3.5	Consistent with design and rest of Tech Spec.
B 3/4 3-5	3/4.3.3.10	Typo.
B 3/4 3-6	3/4.3.3.11	Typo.
B 3/4 4-9	3/4.4.9	Typo.
B 3/4 6-1	3/4.6.1.2	Revised to be consistent with 10CFR50, Appendix J.
B 3/4 6-2	3/4.6.1.6	R.G. 1.35 is not a proposed R.G.
B 3/4 6-3	3/4.6.1.7	Refer to justification for changes [illegible] .7.
B 3/4 7-6	3/4.7.8	Justification with change in attachment.
B 3/4 8-2	3/4.8.1, 3/4.8.2 & 3/4.8.3	Typo.
B 3/4 11-3	3/4 11.2.1	Typos.
5-2	Fig. 5.1-1	Clarification of Figure.
5-9	Table 5.7-1	Typos.
6-3	Fig. 6.2-1	Update organization chart.
6-4	Fig. 6.2-2	Update organization chart.
6-23	6.12.2	Clarification.

SECTION PAGE

~~REPORTING REQUIREMENTS (Continued)~~

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤112.3% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤28.3% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤35.3% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤10^5 cps	≤1.6×10^5 cps
7. Overtemperature ΔT	6.9	2.83	2.26	See Note 1	See Note 2
8. Overpower ΔT	5.5	1.43	1.35	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.7	0.71	2.49	≥~~1900~~ 1875 psig	≥1866 psig
10. Pressurizer Pressure-High	7.5	0.71	2.49	≤2385 psig	≤2400 psig
11. Pressurizer Water Level-High	8.0	2.18	1.96	≤92% of instrument span	≤93.9% of instrument span

*RTP = RATED THERMAL POWER
**Loop design flow = 95,700 gpm

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CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two centrifugal charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

With only one centrifugal charging pump OPERABLE, restore at least two centrifugal charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two centrifugal charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, that the pump develops a differential pressure of greater than or equal to 2400 psid when tested pursuant to Specification 4.0.5.

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours prior to the temperature of one or more of the RCS cold legs exceeding 375°F.

or

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delete

~~Figure 3.1-2 left blank pending NRC approval of three-loop operation~~

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained value of R obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.3.6 The feedwater venturi shall be inspected for fouling and cleaned as necessary at least once per 18 months.

delete

Justification for specification 4.2.3.6, pg. 3/4 2-10:

The WCGS feedwater system uses a venturi as the primary method of flow measurement. WCGS also has the ability to measure flow using the Westinghouse leading edge flow meter (LEFM). The LEFM has a higher accuracy level than the venturi and its design is unaffected by crud buildup. The LEFM can therefore be used to detect crud buildup on the venturi and venturi cleaning can then be scheduled. This is a standard operating use at other LEFM installations.

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3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

   a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

   b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours, and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER. All CAPS

*See Special Test Exception Specification 3.10.2.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##**	4
b. Shutdown	2	1	2	3, 4, 5 3**	5
7. Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6#
8. Overpower ΔT Four Loop Operation	4	2	3	1, 2	6#
9. Pressurizer Pressure-Low	4	2	3	1	6#
10. Pressurizer Pressure-High	4	2	3	1, 2	6#

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be in the closed position and the Control Rod Drive System is capable of rod withdrawal.

**The boron dilution flux doubling signal may be blocked during reactor startup or in accordance with normal operating procedures.

#The provisions of Specification 3.0.4 are not applicable.

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

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

ACTION STATEMENTS

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

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

a. The inoperable channel is placed in the tripped condition within 1 hour;

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1; and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; or

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

Justification for Table 3.3-1, pg. 3/4 3-2 and 3/4 3-5:

This change allows startup which is otherwise prevented with signal not blocked.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

Replace with INSERT.

ACTION 5 - ~~With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers, suspend all operations involving positive reactivity changes and verify Valves BG-V178 and BG-V601 are closed and secured in position within the next hour. With no channels OPERABLE verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, and take the actions stated above within 1 hour and verify compliance at least once per 12 hours thereafter.~~

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour; and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 1 hour.

Insert for Action 5, pg. 3/4 3-6:

Action 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip Breakers, suspend all operations involving positive reactivity changes and verify valves BG-V178 and BG-V601 are closed and secured in position within the next hour.

b. With no channels OPERABLE, open the Reactor Trip Breakers, suspend all operations involving positive reactivity changes and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and every 12 hours thereafter, and verify valves BG-V178 and BG-V601 are closed and secured in position within 4 hours and verified to be closed and secured in position every 14 days.

Justification for Table 3.3-1, Action 5, pg. 3/4 3-6:

Verification time frames have been changed to allow reasonable time to perform actions and to reflect the administrative controls provided by the locked valve list.

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   3.3 – 3

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3.3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $Z + R + S \leq TA$

   Where:

   Z = The value from Column Z of Table 3.3-4 for the affected channel,

   R = The "as measured" value (in percent span) of rack error for the affected channel,

   S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-4 for the affected channel, and

   TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by the performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Loss of Power					
a. 4 kV Undervoltage -Loss of Voltage	N.A.	N.A.	N.A.	83 ± 3 V > 83V (120V Bus) w/1s delay 1.0±0.1s	74.7 to 89V > 74.7V (120V Bus) w/1 + 0.2, -0.5s delay
b. 4 kV Undervoltage -Grid Degraded Voltage	N.A.	N.A.	N.A.	106.9V ≥ 107.1V (120V Bus) w/119s delay# w/8 ± 0.5s delay##	104.3V ≥ 104.5V (120V Bus) w/119 ± 11.6s delay# w/ ≤ 8.5s delay ##
9. Control Room Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	N.A.
d. Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Trip Setpoints and Allowable Values.				
10. Solid-State Load Sequencer	N.A.	N.A.	N.A.	N.A.	N.A.
11. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 1970 psig	≤ 1979 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.

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TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

Applicable to channel calibration only.

Applicable to trip actuating device operational test only.

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq 29^{(1)}/12^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation	$\leq 2^{(5)}$
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 24^{(3)}/12^{(4)}$
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation	$\leq 2^{(5)}$
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling ~~Fans~~	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.
b. Steam Line Isolation	≤ 7

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Purge Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
3) Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	M(1)(2)	N.A.	N.A.	1, 2, 3, 4
4) Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Surveillance Requirements.							

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TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere-Gaseous Radioactivity-High (GT-RE-31 & 32)	1	2	All	###	26
b. Gaseous Radioactivity-RCS Leakage Detection (GT-RE-31 & 32)	N.A.	1	1, 2, 3, 4	N.A.	29
c. Particulate Radioactivity-RCS Leakage Detection (GT-RE-31 & 32)	N.A.	1	1, 2, 3, 4	N.A.	29
2. Fuel Building					
a. Fuel Building Exhaust-Gaseous Radioactivity-High (GG-RE-27 & 28)	1	2	**	##	30
b. Criticality- High Radiation Level					
1) Spent Fuel Pool (SD-RE-37 or 38)	1	1	*	≤ 15 mR/h	28
2) New Fuel Pool (SD-RE-35 or 36)	1	1	*	≤ 15 mR/h	28
3. Control Room					
Air Intake-Gaseous Radioactivity-High (GK-RE-04 & 05)	1	2	All	#	27

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure		
a) Normal Range	2	1
b) Extended Range	2	1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Containment Hydrogen Concentration Level	2	1
11. Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
12. PORV Position Indicator*	1/Valve	1/Valve
13. PORV Block Valve Position Indicator**	1/Valve	1/Valve
14. Safety Valve Position Indicator	1/Valve	1/Valve
15. Containment Water Level	2	1
16. Containment Radiation Level (High Range)	N.A.	1
17. Thermocouple/Core Cooling Detection System	4/core quadrant	2/core quadrant
18. Unit Vent - High Range~~s~~ Noble Gas Monitor	N.A.	1

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position and power is removed.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Containment Hydrogen Concentration Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. PORV Position Indicator*	M	N.A.
13. PORV Block Valve Position Indicator**	M	N.A.
14. Safety Valve Position Indicator	M	N.A.
15. Containment Water Level	M	R
16. Containment Radiation Level (High Range)	M	R***
17. Thermocouple/Core Cooling Detection System	M	R
18. Unit Vent - High Range Noble Gas Monitor	M	R

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position and power is removed.

***CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point ~~calibution~~ Calibration check of the detector below 10 R/h with an installed or portable gamma source.

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
1101-Aux. Bldg. 1974' Gen. Flr. #1	100			0/11
1102-Chiller & Surge Tks. Area	100			0/4
1102-Chiller & Surge Tks. Area	101			2/0
1107-Cent. Charg. Pmp. Rm. B	101			2/0
1108-Safety Inj. Pmp. Rm. B	101			2/0
1109-Res. Ht. Remov. Pmp. Rm. B	101			1/0
1110-Ctmt. Spray Pmp. Rm. B	101			1/0
1111-Res. Ht. Remov. Pmp. Rm. A	101			1/0
1112-Ctmt. Spray Pmp. Rm. A	101			1/0
1113-Safety Inj. Pmp. Rm. A	101			2/0
1114-Cent. Charg. Pmp. Rm. A	101			2/0
1115-Pos. Disp. Charg. Pmp. Rm.	101			2/0
1116, 1117-Boric Acid Tk. Rms.	101			2/0
1116, 1117-Boric Acid Tk. Rms.	101		2/0	
1120-Aux. Bldg. 1974' Gen. Flr. #2	101			4/0
1122-Aux. Bldg. 1974' Gen. Flr. #3	100			0/3
1122-Aux. Bldg. 1974' Gen. Flr. #3	101			5/0
1126-Boron Inj. Tk. & Pmp. Rm.	101			1/0
1127-Stair A-Z	109			1/0
1128- Aux. Feedwater Pump Rm. Basement	117			2/0
1130-Aux. Bldg. 1974' N. Corr.	100			0/2
1206-W. Pipe Chase Below AFWP Area	117			2/0
1203-Aux. Bldg. Elec. Chase S. 1988'	117			1/0
1301-Aux. Bldg. 2000' Corridor #1	103			0/10
1301-Aux. Bldg. 2000' Corridor #1	117			2/0
1311-Aux. Bldg. Sampling Rm.	117			2/0
1312-Boron Meter/RC Activity Mon. Rm.	103			0/1
1314-Aux. Bldg. 2000' Corridor #3	103			0/3
1314-Aux. Bldg. 2000' Corridor #3	117			2/0
1315-Ctmt. Spray Add. Tk. Area	103			0/2
1316-Vlv. Rm. by Seal Wtr. Ht. Exch.	103			0/1
1320-Aux. Bldg. 2000' Corridor #4	103			0/3
1321-Aux. Bldg. 2000' S. Exit Vest.	103			0/1
1322-Pipe Pene. Rm. B	117			5/0
1323-Pipe Pene. Rm. A	117			6/0
1325-Aux. FW Pmp. Rm. B	117			2/0
1326-Aux. FW Pmp. Rm. A	117			2/0
1331-Aux. FW Pmp. Rm. C	111		2/0	
1331-Aux. FW Pmp. Rm. C	117			1/0
1335-Aux. Bldg. Elec. Chase N. 2000'	117			1/0
1336-Aux. Bldg. Elec. Chase S. 2000'	117			1/0
1401-Comp. Cool. Pmp. & Ht. Exch. B	118			5/0
1402-Aux. Bldg. 2026' Corridor #1	104			0/6
1403-MG Set Rm.	105			0/9(1)
1403-MG Set Rm.	112			0/9(1)

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
1405-Chemical Stg. Area	118			6/0
1406-Comp. Cool. Pmp. & Ht. Exch. A	104			0/1
1406-Comp. Cool. Pmp. & Ht. Exch. A	118			2/0
1408-Aux. Bldg. 2026' Corridor #2	104			0/9
1408-Aux. Bldg. 2026' Corridor #2	118			5/0
1409-Elec. Pene. Rm. B	106			0/4(1)
1409-Elec. Pene. Rm. B	113			0/4(1)
1410-Elec. Pene. Rm. A	107			0/8(1)
1410-Elec. Pene. Rm. A	114			0/8(1)
1413-Aux. Shutdown Pnl. Rm.	118			4/0
1501-Ctrl. Rm. A/C & Filt. Units B	110			10/0
1504-Ctmt. Purge Exh. & Mech. Equip. B	108			18/0
1506-Cmt. Purge Sup. AHU Rm. A	109			18/0
1507-Personnel Hatch Area	108			3/0
1508-Main Steam Iso. Valve Rm #1	115		1/0	
1509-Main Steam Iso. Valve Rm #2	115		1/0	
1512-Ctrl. Rm. A/C & Filt. Units A	110			10/0
1513-Ctrl. Bldg. Vent Sup. A/C Unit Rm.	109			3/0
1513-Aux. Bldg. Duct 2047'6"	119			1/0
NA - Containment**	201	1/0(2)		
NA - Containment**	202	2/0(2)		
NA - Containment**	203	1/0(2)		
NA - Containment**	204	1/0(2)		
NA - Containment**	206	3/0(2)		
NA - Containment**	215	1/0(2)		
NA - Containment**	216	1/0(2)		
NA - Containment**	217	1/0(2)		
NA - Containment**	218	1/0(2)		
NA - Containment**	219			4/0
NA - Containment**	220	1/0(2)		
3101-Ctrl. Bldg. 1974' Pipe Space	300			11/0
3105-Ctrl. Bldg. Elec. Chase S. 1974'	300			1/0
3106-Ctrl. Bldg. Elec. Chase N. 1974'	300			1/0
NA -Area Above Access Control	301			12/0
3229-Ctrl. Bldg. Elec. Chase S. 1984'	300			1/0
3230-Ctrl. Bldg. Elec. Chase N. 1984'	300			1/0
3301-ESF Swgr. Rm. #1	314			0/7(1)
3301-ESF Swgr. Rm. #1	315			0/7(1)
3302-ESF Swgr. Rm. #2	316			0/5(1)
3302-ESF Swgr. Rm. #2	317			0/5(1)
3305-Ctrl. Bldg. Elec. Chase S. 2000'	301			1/0
3306-Ctrl. Bldg. Elec. Chase N. 2000'	301			1/0
3403-Non-Vit. Swgr. & Xfmr. Rm. #1	304			0/1(1)
3403-Non-Vit. Swgr. & Xfmr. Rm. #1	305			0/1(1)
3404-Switchboard Rm. #4	321			0/2(1)

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
3404-Switchboard Rm. #4	322			0/2(1)
3405-Battery Rm. #4	303			2/0
3407-Battery Rm. #1	303			2/0
3408-Switchboard Rm. #1	325			0/2(1)
3408-Switchboard Rm. #1	326			0/2(1)
3409-Non-Vit. Swgr. & Xfmr. Rm. #2	323			0/1(1)
3409-Non-Vit. Swgr. & Xfmr. Rm. #2	327			0/1(1)
3410-Switchboard Rm. #2	324			0/2(1)
3410-Switchboard Rm. #2	328			0/2(1)
3411-Battery Rm. #2	303			2/0
3413-Battery Rm. #3	303			1/0
3414-Switchboard Rm. #3	318			0/2(1)
3414-Switchboard Rm. #3	320			0/2(1)
3415-Acc. Ctrl. & Elec. Equip. A/C Units #1	303			4/0
3416-Acc. Ctrl. & Elec. Equip. A/C Units #2	303			4/0
3418-Ctrl. Bldg. Elec. Chase S. 2016'	303			1/0
~~3419~~ 3414 -Ctrl. Bldg. Elec. Chase N. 2016'	303			1/0
3414 -Ctrl. Bldg. Elec. Chase N. 2016'	303			1/0
3410 -Ctrl. Bldg. Elec. Chase S. 2016'	303			1/0
3501-Lower Cable Spreading Rm.	306			0/13
3504-Ctrl. Bldg. Elec. Chase N. 2032'	303			1/0
3505-Ctrl. Bldg. Elec. Chase S. 2032'	303			1/0
3501 -Ctrl. Bldg. Elec. Chase N. 2032'	303			1/0
3501 -Ctrl. Bldg. Elec. Chase S. 2032'	303			1/0
3601-Control Room	308			4/0
3601-Control Room	309			0/7(1)
3601-Control Room	319			0/7(1)
3601-Control Room	329			20/0
3602-Pantry	308			1/0
3603-Shift Supv. Office	308			1/0
3605-Equipment Cabinet Area	308			15/0
3606-Emerg. Equip. Storage Rm.	308			1/0
3608-Janitor's Closet	308			1/0
3609-SAS Rm.	308			1/0
3617-Ctrl. Bldg. Elec. Chase S. 2047'6"	308			1/0
3618-Ctrl. Bldg. Elec. Chase N. 2047'6"	308			1/0
3605 -Ctrl. Bldg. Elec. Chase S. 2047'6"	308			1/0
3801-Upper Cable Spreading Rm.	307			0/18
3804-Ctrl. Bldg. Elec. Chase S. 2073'6"	308			1/0
3801-Ctrl. Bldg. Elec. Chase S. 2073'6"	308			1/0
5201-W. Diesel Gen. Rm.	501		4/0	
5201-W. Diesel Gen. Rm.	502	0/8		
5203-E. Diesel Gen. Rm.	500		4/0	

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TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	ZONE	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
5203-E. Diesel Gen. Rm.	503	0/8		
6102-Fuel Bldg. Railroad Bay	600	0/8		
6104-Fuel Pool Cool. HX Rm. B	601			6/0
6105-Fuel Pool Cool. HX Rm. A	601			6/0
6202-Elec. Equipment Rm.	601			3/0
6203-Air Handling Equip. Rm.	601			3/0
6301-Fuel Bldg. 2047'6" Gen. Flr.	602		2/0	
6303-Fuel Bldg. Exh. Filt. Absorb. Rm. A	601			2/0
6304-Fuel Bldg. Exh. Filt. Absorb. Rm. B	601			2/0
NA -ESW Pumphouse Train B	002			3/0
NA -ESW Pumphouse Train A	001			3/0
NA -ESF Transformer XNB01	016	0/6		
NA -ESF Transformer XNB02	017	0/6		

TABLE NOTATIONS

*(x/y): x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

**The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

(1) Zone is associated with a Halon-protected space. Each space has two separate detection circuits (zones). One zone, in its entirety, needs to remain OPERABLE.

(2) Line-type heat detector.

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TABLE 3.3-13 (Continued)

TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

ACTION STATEMENTS

ACTION 38 - 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 39 - 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 40 - 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 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 41 - 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 42 - With the Outlet Oxygen Monitor channel inoperable, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both oxygen channels or both the inlet oxygen and inlet hydrogen channels inoperable, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

ACTION 43 - 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 sample equipment as required in Table 4.11-2.

ACTION 44 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, suspended oxygen supply to the recombiner.

ACTION 45 - Flow rate for this system shall be based on fan status and operating curves or actual measurements.

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TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* At all times.

** During WASTE GAS HOLDUP SYSTEM operation.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation as appropriate occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint (isolation and alarm), or

b. Circuit failure (alarm only), or

c. Instrument indicates a downscale failure (alarm only) or

d. Instrument controls not set in operate mode (alarm only).

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that contpol room alarm annunciation occurs if any of the following conditions exist~~s~~:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure, or

d. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference (gas or liquid and solid) standards certified by the National Bureau of Standards (NBS) 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, measurement range, and establish monitor response to a solid calibration source. For subsequent CHANNEL CALIBRATION, NBS traceable standard (gas, liquid, or solid) may be used; or a gas, liquid, or solid source that has been calibrated by relating it to equipment that was previously (within 30 days) calibrated by the same geometry and type of source traceable to NBS.

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

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

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REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least three of the reactor coolant loops listed below shall be OPERABLE and at least two of these reactor coolant loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop D and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.**

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one reactor coolant loop in operation, restore at least two loops to operation within 72 hours or within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side wide range water level to be greater than or equal to 10% at least once per 12 hours.

4.4.1.2.3 At least two reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

---

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**See Special Exception Specification 3.10.4.

Test

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3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With one or more PORV(s) inoperable because of excessive ~~seat~~ leakage, within 1 hour either restore the PORV(s) to OPERABLE ~~status or~~ close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one PORV inoperable due to causes other than excessive ~~seat~~ leakage, within 1 hour either restore the PORV to OPERABLE ~~status~~, or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With both PORV(s) inoperable due to causes other than excessive ~~seat~~ leakage, within 1 hour either restore each of the PORV(s) to ~~OPERABLE~~ status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV to remove power from its associated solenoid valve; and (2) apply ACTION b. or c. above, as appropriate, for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

*With all RCS cold leg temperatures above 368°F.

Justification for specification 3.4.4, Actions a., b., and c., pg. 3/4 4-10:

Other types of leakages develop in these valves (such as body-to-bonnet) which are just as isoable by the block valves as seat leakage. The intent of this specification is to allow continual operation if the PORV is operable except for relatively small leakages. Deletion of the adverb "seat" allows this intent to be realized.

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REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two residual heat removal (RHR) suction relief valves each with a Setpoint of 450 psig ± 3%, or

b. Two power-operated relief valves (PORVs) with Setpoints which do not exceed the limit established in Figure 3.4-4, or

c. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2 square inches.

APPLICABILITY: MODE 3 when the temperature of any RCS cold leg is less than or equal to 368°F, MODES 4 and 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With ~~less than two~~ ONE PORV or ~~two~~ ONE RHR suction relief valve~~s~~ inoperable ~~OPERABLE~~, either restore two PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With both PORVs and both RHR suction relief valves inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

c. In the event the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

CONTAINMENT SYSTEMS

FINAL DRAFT

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, 0.20% by weight of the containment air per 24 hours at P_a, 48 psig, or

2) Less than or equal to L_t, ___% by weight of the containment air per 24 hours at P_t, ~~24 psig~~.

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, 48 psig.

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

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a or less than L_t, as applicable, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than either P_a, 48 psig, or P_t, 24 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

Justification for specification 3.6.1.2a.2), pg. 3/4 6-2:

This information will be provided after testing has been accomplished. (ILRT is currently scheduled for Dec. 15. Data available by Dec. 25.)

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valves shall be OPERABLE and:

a. Each 36-inch containment shutdown purge supply and exhaust isolation valve shall be closed and blank flanged, and

b. The 18-inch containment mini-purge supply and exhaust isolation valve(s) may be open for up to ~~500~~ 2000 hours during a calendar year.

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

ACTION:

a. With a 36-inch containment purge supply and/or exhaust isolation valve open or not blank flanged, close and/or blank flange that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the 18-inch containment mini-purge supply and/or exhaust isolation valve(s) open for more than ~~500~~ 2000 hours during a calendar year, close the open 18-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.7.2 and/or 4.6.1.7.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

Justification for Specification 3.6.1.7b. and Action b.:

KG&E feels 2000 hours of allowed purge time in MODES 1 through 4 versus the Standard Technical Specification of 500 hours is justified based on safety, operational and maintenance considerations.

SAFETY

The containment isolation valves in the system were selected, tested, and located in accordance with the requirements of 10 CFR 50, Appendix A, General Design Criteria 54 and 56, and 10 CFR 50, Appendix J, Type C testing.

The actuation circuitry for Containment Purge Isolation is Safety Grade, Class IE. Isolation is initiated by any Safety Injection Signal, High Containment Airborne Activity, or High Containment Purge Exhaust Activity.

OPERATION AND MAINTENANCE

KG&E expects that during the first year of commercial operation the number of pre-planned entries and the number of unplanned entries into containment would be higher than in subsequent years. Good engineering practice dictates more frequent observation and surveillance activities for a new plant. In addition, the Standard Technical Specifications mandate special surveillance of snubbers during the first fuel cycle. The "shakedown" phase of the plant will normally result in more forced outages requiring unplanned entries into containment.

The increased purge time would reduce radiation levels within containment and therefore be consistent with an ALARA philosophy.

CONCLUSION

The change poses no increased threat to the health and safety of the public and will allow KG&E to operate the Wolf Creek Plant within the philosophies of good engineering practices and ALARA.

FINAL DRAFT

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION (Seconds)
1. Phase "A" Isolation (active) - (Continued)				
P-32	LF FV-96	CTMT Normal Sumps to Floor Drain Tank Outside CTMT Iso	C	4
P-93	SJ HV-5**	PZR/RCS Liquid Sample Inner CTMT Iso	C	5
P-93	SJ HV-6**	PZR/RCS Liquid Sample Outer CTMT Iso	C	5
P-69	SJ HV-12**	PZR Vapor Sample Inner ~~Outer~~ CTMT Iso	C	5
P-69	SJ HV-13**	PZR Vapor Sample Outer ~~Inner~~ CTMT Iso	C	5
P-95	SJ HV-18**	Accumulator Sample Inner CTMT Iso	C	5
P-95	SJ HV-19**	Accumulator Sample Outer CTMT Iso	C	5
P-93	SJ HV-127**	PZR/RCS Liquid Sample Outer CTMT Iso	C	5
P-64	SJ HV-128**	PZR/RCS Liquid Sample Inner CTMT Iso	A,C	5
P-64	SJ HV-129**	PZR/RCS Liquid Sample Outer CTMT Iso	A,C	5
P-64	SJ HV-130**	PZR/RCS Liquid Sample Outer CTMT Iso Valve	A,C	5
P-57	SJ HV-131**	PASS Discharge to RCDT	A,C	5
P-57	SJ HV-132**	PASS Discharge to RCDT	A,C	5
2. Phase "A" Isolation (passive)*				
P-58	EM HV-8888**	Accumulator Tank Fill Line Iso Valve	C	5

*May be opened on an intermittent basis under administrative control.
**The provisions of Specification 3.0.4 are not applicable.

FINAL DRAFT

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION (Seconds)
6. Remote Manual - (Continued)				
P-29	EF HV-48	ESW Return From Containment Coolers	C	N.A.
P-73	EF HV-49	ESW Return From Containment Coolers	C	N.A.
P-29	EF HV-50	ESW Return From Containment Coolers	C	N.A.
P-74	EG HV-127*	CCW Supply to RCP	C	N.A.
P-75	EG HV-130*	CCW Return from RCP	C	N.A.
P-75	EG HV-131*	CCW Return From RCP	C	N.A.
P-76	EG HV-132*	CCW Return From RCP Thermal Barriers	C	N.A.
P-76	EG HV-133*	CCW Return from RCP Thermal Barrier	C	N.A.
P-79	EJ HV-8701A	RCS Hot Leg 1 to RHR Pump A Suction	A	N.A.
P-52	EJ HV-8701B	RCS Hot Leg 4 to RHR Pump B Suction	A	N.A.
P-82	EJ HV-8809A	RHR Pump A Cold Leg Injection Iso Valve	A	N.A.
P-27	EJ HV-8809B	RHR Pump B Cold Leg Injection Iso Valve	A	N.A.
P-15	EJ HV-8811A	CTMT Recirc Sump to RHR Pump A Suction	A	N.A.

*These valves were assumed to be closed during the accident analysis, and are normally closed but may be opened on an intermittant basis under adminstrative control.

FINAL DRAFT

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TI (Seconds)
8. Hand-Operated and Check Valves - (Continued)				
P-24	BG V-135	RCP Seal Water Return	C	N.A.
P-80	BG 8381	CVCS Charging Line	C	N.A.
P-25	BL 8046	Reactor Makeup Water Supply	C	N.A.
P-78	BM V-045	Steam Generator Drain Line Iso Valve	C	N.A.
P-78	BM V-046	Steam Generator Drain Line Iso Valve	C	N.A.
P-53	EC V-083	Refueling Pool Supply From Fuel Pool Cleanup	C	N.A.
P-53	EC V-084	Refueling Pool Supply From Fuel Pool Cleanup	C	N.A.
P-54	EC V-087	Refueling Pool Return to Fuel Pool Cooling	C	N.A.
P-54	EC V-088	Refueling Pool Return to Fuel Pool Cooling	C	N.A.
P-55	EC V-095	Refueling Pool Skimmers To Fuel Pool Cooling Loop	C	N.A.
P-55	EC V-096	Refueling Pool Skimmers To Fuel Pool Cooling Loop	C	N.A.
P-74	EG V-204	CCW Supply to RCP	C	N.A.
P-82	EP ~~EJ~~ 8818A	RHR Pump to Cold Leg 1 Injection	A	N.A.
P-82	EP ~~EJ~~ 8818B	RHR Pump to Cold Leg 2 Injection	A	N.A.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

8. Hand-Operated and Check Valves - (Continued)

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TI (Seconds)
P-27	EP ~~EJ~~ 8818C	RHR Pump to Cold Leg 3 Injection	A	N.A.
P-27	EP ~~EJ~~ 8818D	RHR Pump to Cold Leg 4 Injection	A	N.A.
P-21	EJ 8841A	RHR Pump Disch to RCS Hot Leg 2	A	N.A.
P-21	EJ 8841B	RHR Pump Disch to RCS Hot Leg 3	A	N.A.
P-87	EM V-001	SI Pump Hot Leg 1 Injection	A	N.A.
P-87	EM V-002	SI Pump Hot Leg 2 Injection	A	N.A.
P-48	EM V-003	SI Pump Hot Leg 3 Injection	A	N.A.
P-48	EM V-004	SI Pump Hot Leg 4 Injection	A	N.A.
P-58	EM V-006	Accumulator Fill Line From SI Pumps	C	N.A.
P-49	EM V-010	SI Pump Disch to Cold Leg 1	A	N.A.
P-49	EM V-020	SI Pump Disch to Cold Leg 2	A	N.A.
P-49	EM V-030	SI Pump Disch to Cold Leg 3	A	N.A.
P-49	EM V-040	SI Pump Disch to Cold Leg 4	A	N.A.
P-88	EM V-8815	BIT to RCS Cold Leg Injection	A	N.A.
P-89	EN V-013	CTMT Spray Pump A to CTMT Spray Nozzles	A	N.A.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION (Seconds)
8. Hand-Operated and Check Valves - (Continued)				
P-66	EN V-017	CTMT Spray Pump B to CTMT Spray Nozzles	A	N.A.
P-45	EP V-046	Accumulator Nitrogen Supply Line	C	N.A.
P-43	HD V-016	Auxiliary Steam to Decon System	C	N.A.
P-43	HD V-017	Auxiliary Steam to Decon System	C	N.A.
P-63	KA V-039	Rx Bldg Service Air Supply	C	N.A.
P-63	KA V-118	Rx Bldg Service Air Supply	C	N.A.
P-98	KB V-001	Breathing Air Supply to RX Bldg	C	N.A.
P-98	KB V-002	Breathing Air Supply to RX Bldg	C	N.A.
P-30	KA V-204	Rx Bldg Instrument Air Supply	C	N.A.
P-67	KC V-478	Fire Protection Supply to RX Bldg	C	N.A.
P-57	SJ V-111	Liquid Sample from PASS to RCDT	A,C	N.A.
9. Other Automatic Valves				
~~P-1~~	~~AB-HV-11**~~	~~Mn. Stm. Isol.~~	~~A~~	~~5~~
~~P-2~~	~~AB-HV-14**~~	~~Mn. Stm. Isol.~~	~~A~~	~~5~~
~~P-3~~	~~AB-HV-17**~~	~~Mn. Stm. Isol.~~	~~A~~	~~5~~

**The provisions of Specification 3.0.4 are not applicable.

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION T (Seconds)
9. Other Automatic Valves (Continued)				
~~P-4~~	~~AB-HV-20**~~	~~Mn. Stm. Isol.~~	~~A~~	~~5~~
~~P-5~~	~~AE-FV-42**~~	~~Mn. FW Isol.~~	~~A~~	~~5~~
~~P-6~~	~~AE-FV-39**~~	~~Mn. FW Isol.~~	~~A~~	~~5~~
~~P-7~~	~~AE-FV-40**~~	~~Mn. FW Isol.~~	~~A~~	~~5~~
~~P-8~~	~~AE-FV-41**~~	~~Mn. FW Isol.~~	~~A~~	~~5~~
P-9	BM-HV-4**	SG Blowdn. Isol.	A	10
P-10	BM-HV-1**	SG Blowdn. Isol.	A	10
P-11	BM-HV-2**	SG Blowdn. Isol.	A	10
P-12	BM-HV-3**	SG Blowdn. Isol.	A	10

**The provisions of Specification 3.0.4 are not applicable.

Justification for Table 3.6-1 9, pg. 3/4 6-29 and 3/4 6-30:

These valves are not considered containment isolation valves in the FSAR, pg. 6.2.4-6 and Section 15.6.3.2. The response time testing for these valves is contained in Specification Table 3.3-5 (steam line isolation testing and feedwater isolation testing).

d. At least once per 6 months by performance of a yard loop and fire hydrant flush,

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

~~1) Verifying that each automatic valve in the flow path actuates to its correct position,~~

1~~2~~) Verifying that each pump velops at least 3300 gpm at a system pressure of 125 psi, 80 psig

2~~3~~) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

3~~4~~) ~~Verifying that each fire suppression pump starts (sequentially) on decreasing pressure in the fire suppression header at a header pressure greater than or equal to 80 psig.~~

Replace with Insert

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

4.7.10.1.2 Each fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel storage tank contains at least 200 gallons of fuel, and

2) The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975, is within the acceptable limits specified in Table 1 of ASTM D975-1977 when checked for viscosity, water, and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

Insert for specification 4.7.10.1.1 3), pg. 3/4 7-28:

Verifying that the electric driven fire pump starts on a start signal initiated on decreasing header pressure of 75 psig and the diesel driven fire pump starts on a start signal on decreasing header pressure of 70 psig after 10 second time delay to avoid simultaneous start of both pumps.

TABLE 3.7-3

FIRE HOSE STATIONS

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BUILDING	ELEVATION	AREA	HOSE RACK
Auxiliary	1974	1122	KC-HR-051
Auxiliary	1974	1122	KC-HR-047
Auxiliary	1974	1120	KC-HR-031
Auxiliary	1974	1120	KC-HR-025#
Auxiliary	1974	1101	KC-HR-023#
Auxiliary	1974	1101	KC-HR-040
Auxiliary	1974	1101	KC-HR-042
Auxiliary	1988	1201	KC-HR-024
Auxiliary	2000	1329	KC-HR-111
Auxiliary	2000	1320	KC-HR-048
Auxiliary	2000	1320	KC-HR-046#
Auxiliary	2000	1314	KC-HR-030
Auxiliary	2000	1321	KC-HR-029#
Auxiliary	2000	1301	KC-HR-035#
Auxiliary	2000	1301	KC-HR-039
Auxiliary	2000	1301	KC-HR-041#
Auxiliary	2026	1408	KC-HR-049
Auxiliary	2026	1408	KC-HR-044
Auxiliary	2026	1408	KC-HR-032#
Auxiliary	2026	1408	KC-HR-026#
Auxiliary	2026	1401	KC-HR-034
Auxiliary	2026	1403	KC-HR-037#
Auxiliary	2047	1506	KC-HR-050
Auxiliary	2047	1513	KC-HR-043
Auxiliary	2047	1506	KC-HR-045
Auxiliary	2047	1501	KC-HR-038
Auxiliary	2047	1504	KC-HR-033
Auxiliary	2047	1502	KC-HR-027
Auxiliary	2064	1119	KC-HR-028#
Control	1974	3101	KCHR-002# KC-HR-002#
Control	1974	3101	KC-HR-014#
Control	1984	3204	KC-HR-015#
Control	1984	3221	KC-HR-001#
Control	2000	3301	KC-HR-004#
Control	2000	3301	KC-HR-017#
Control	2000	3302	KC-HR-016#
Control	2016	3401	KC-HR-005
Control	2016	3401	KC-HR-019
Control	2016	3401	KC-HR-018

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TABLE 3.7-3 (Continued)

FIRE HOSE STATIONS

BUILDING	ELEVATION	AREA	HOSE RACK
Control	2032	3501	KC-HR-006#
Control	2032	3501	KC-HR-020#
Control	2047	3604	KC-HR-007
Control	2047	3616	KC-HR-021
Control	2073	3801	KC-HR-008#
Control	2073	3801	KC-HR-022#
Reactor	2000	2201	KC-HR-120*
Reactor	2000	2201	KC-HR-131*
Reactor	2000	2201	KC-HR-124*
Reactor	2000	2201	KC-HR-129*
Reactor	2026	N.A.	KC-HR-121*
Reactor	2026	N.A.	KC-HR-132* #
Reactor	2026	N.A.	KC-HR-125*
Reactor	2026	N.A.	KC-HR-130*
Reactor	2047	N.A.	KC-HR-128*
Reactor	2047	N.A.	KC-HR-122*
Reactor	2047	N.A.	KC-HR-126*
Reactor	2068	N.A.	KC-HR-123*
Reactor	2068	N.A.	KC-HR-127*
Fuel	2000	6102	KC-HR-142#
Fuel	2000	6102	KC-HR-054#
Fuel	2000	6102	KC-HR-143
Fuel	2000	6104	KC-HR-057
Fuel	2026	6201	KC-HR-133
Fuel	2026	6203	KC-HR-052
Fuel	2047	6301	KC-HR-055#
Fuel	2047	6302	KC-HR-056#
Fuel	2047	6301	KC-HR-053#
ESW	2000	N.A.	KC-HR-140
ESW	2000	N.A.	KC-HR-141

TABLF NOTATIONS

#Secondary means of fire suppression to Water Sprays/Deluge or Halon Systems.
*Fire hose for station to be stored external to Reactor Building.

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3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

4160 +160 -480 volts

4) Verifying the diesel starts from ambient condition and accelerates to at least 514 rpm in less than or equal to 12 seconds.* The generator voltage and frequency shall be ~~4000 ± 320 volts~~ and 60 ± 1.2 Hz within 12 seconds* after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

c) Safety Injection test signal.

5) Verifying the generator is synchronized, loaded to greater than or equal to 6201 kW in less than or equal to 60 seconds,* operates with a load greater than or equal to 6201 kW for at least 60 minutes, and

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks;

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

d. By sampling new fuel oil in accordance with ASTM D4057 prior to addition to storage tanks and:

(1) By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the storage tanks that the sample has:

(a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

*These diesel generator starts from ambient conditions shall be performed only once per 184 days in these surveillance tests and all other engine starts for the purpose of this surveillance testing shall be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

(b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification;

(c) A flash point equal to or greater than 125°F; and

(d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

(2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

e. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-78, Method A.

f. At least once per 18 months, during shutdown, by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

2) Verifying the diesel generator capability to reject a load of greater ~~than or equal to~~ 1352 kW (ESW pump) while maintaining voltage at ~~4000 ± 320~~ 4160 +160 -480 volts and frequency at 60 ± 5.4 Hz,

3) Verifying the diesel generator capability to reject a load of 6201 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection,

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 12 seconds, energizes the auto-connected shutdown loads through the shutdown sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at ~~4000 ± 320~~ volts and 60 ± 1.2 Hz during this test. 4160 +160 - 480

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5) Verifying that on a Safety Injection test signal without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes; and the offsite power source energizes the auto-connected emergency (accident) load through the LOCA sequencer. The generator voltage and frequency shall be ~~4000 ± 320~~ volts and 60 ± 1.2 Hz within 12 seconds after the auto-start signal; the generator steady-state generator voltage and frequency shall be maintained within these limits during this test; 4160 + 160 - 480

6) Simulating a loss-of-offsite power in conjunction with a Safety Injection test signal, and

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 12 seconds, energizes the auto-connected emergency (accident) loads through the LOCA sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at ~~4000 ± 320~~ volts and 60 ± 1.2 Hz during this test; and 4160 + 160 - 480

c) Verifying that all automatic diesel generator trips, except high jacket coolant temperature, engine overspeed, low lube oil pressure, high crankcase pressure, start failure relay, and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 6821 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 6201 kW. The generator voltage and frequency shall be ~~4000 ± 320~~ volts and 60 + 1.2 Hz, - 3 Hz within 12 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within 4160 + 160 - 480 ~~4000 ± 320~~ volts and 60 ± 1.2 Hz during this test. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2f.6)b)*; 4160 + 160 - 480

*If Specification 4.8.1.1.2f.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at 6201 kW for 1 hour or until operating temperature has stabilized.

Justification for specification 4.8.1.1.2 a. 4), f. 2), f. 4) b), f. 5), f. 6)b), and f. 7). pg. 3/4 8-3,4, and 5:

Shifted center point but maintained span at same points. Change needed to ensure compatability with specifications 3.8.3.1 and 3.8.3.2. Also the generator is a 4160 volt machine and the regulator is set to regulate at 4160 volts.

FINAL DRAFT

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

1) A day tank containing a minimum volume of 390 gallons of fuel,

2) A fuel storage system containing a minimum volume of 85,300 gallons of fuel, and

3) A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the spent fuel pool. ~~and within 8 hours, depressurize and vent the Reactor Coolant System through at least a 2 square inch vent.~~ In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5)), and 4.8.1.1.3.

Justification for specification 3.8.1.2 Action, pg. 3/4 8-8:

This should have been deleted when the specifications were modified for use of RHR cold overpressure protection.

FINAL DRAFT

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

a. 125-Volt Battery Bank NK11 and NK13, and its associated Full Capacity Chargers NK21 and NK23, ~~or~~ and

b. 125-Volt Battery Bank NK12 and NK14, and its associated Full Capacity Chargers NK22 and NK24.

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

ACTION:

With one of the required battery banks and/or full capacity chargers inoperable, restore the inoperable battery bank and/or full capacity charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to 130.2 volts on float charge.

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TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE(3) VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and < ¼" above maximum level indication mark	>Minimum level indication mark, and < ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts(6)	> 2.07 volts
Specific Gravity(4)	$\geq$ 1.200(5)	$\geq$ 1.195	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells $\geq$ 1.195(5)

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than 2 amps when on charge.

(6) Corrected for average electrolyte temperature.

FINAL DRAFT

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
6EPK05E P-3A Fuse RL018 B-3A Fuse	Accumulator Water FIll Vlv EPHV8878B
6EPK05F P-3A Fuse RL018 B-3A Fuse	Accumulator Water Fill Vlv EPHV8878D
P-4SJY01B 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Press. Vapor. Cont. Iso. Space Vlv. SJHV12
P-4SJY01C 3A Fuse RL011 B-4RLY01G 15A Breaker NG02ACR140	Accums Sample Cont Isol Vlv SJHV18
P-5SJY03B 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV16
P-5SJY03C 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV17
P-5SJY04B 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV14
P-5SJ0Y4C 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Accumulator Sample Line Vlv SJHV15
P-1SJY06B 3A Fuse RP332 B-1RPY09F 15A Breaker NG01BAR140	HL Sample 3 Vlv SJHV4
P-4SJY06A 3A Fuse RP333 B-4RPY09F 15A Breaker NG02BAR140	HL Sample 1 Vlv SJHV3

Lower case

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TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	POWERED EQUIPMENT
Low Voltage Power and Control (Continued)	
P-5SJY06C 3A Fuse RP211 B-5RPY09D 15A Breaker PG19NHF236	Press Liquid Space Samp Isol Vlv SJHV20
P-4BMY01A 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G. A Out to Nuc Sample Sys Vlv BMHV19
P-4BMY01B 3A Fuse RL024 B-4RLY01H 15A Breaker NG2ACR127	S.G. B Out to Nuc Sample Sys Vlv BMHV20
P-4BMY01C 3A Fuse RL024 B-4RLY01H 15A Breaker NG02ACR127	S.G. C Out to Nuc Sample Sys Vlv BMHV21
P-5GNY08A 3A Fuse RL020 B-5RLY01L 15A Breaker PG19GCR230	CRDM Cooling Discharge Damper GNHZ71
P-5GNY08C 3A Fuse RL020 B-5RLY01L 15A Breaker PG19GCR230	CRDM Cooling Discharge Damper GNHZ73
P-6GNY08A 3A Fuse RL020 B-6RLY01J 15A Breaker PG20GBR222	CRDM Cooling Discharge Damper GNHZ72
P-6GNY08C 3A Fuse RL020 B-6RLY01J 15A Breaker PG20GBR222	CRDM Cooling Discharge Damper GNHZ74
5BGK10A P-3A Fuse RL001 B-3A Fuse	Normal Letdown Isolation Vlv BGLCV459

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revisison 2, March 1978, for a methyl iodide penetration of less than 1%; and

3) Verifying a system flow rate of 9000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 5.4 inches Water Gauge while operating the system at a flow rate of 9000 cfm ± 10%.

2) Verifying that on a Spent Fuel Pool Gaseous Radioactivity-High test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks and isolates the normal fuel building exhaust flow to the auxiliary/fuel building exhaust fan;

3) Verifying that the system maintains the Fuel Building at a negative pressure of greater than or equal to 1/4 inches Water Gauge relative to the outside atmosphere during system operation; and

4) Verifying that the heaters dissipate 37 ± 3 kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of 9000 cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 9000 cfm ± 10%.

FINAL DRAFT

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of the following requirements may be suspended:

a. Specification 3.4.1.1 - During the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

1) The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

2) The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

b. Specification 3.4.1.2 - During the performance of hot rod drop time measurements in MODE 3 provided at least three reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During operation below the P-7 Interlock Setpoint or performance of hot rod drop time measurements.

ACTION:

a. With the THERMAL POWER greater than the P-7 Interlock Setpoint during the performance of startup and PHYSICS TESTS, immediately open the Reactor trip breakers.

b. With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately place two reactor coolant loops in operation.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

4.10.4.3 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of the hot rod drop time measurements and at least once per 4 hours during the hot rod drop time measurements by verifying correct breaker~~s~~ alignments and indicated power availability.

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TABLE 4.11-1 (Continued)

TABLE NOTATIONS

(1)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:

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

Where:

LLD = the "a priori" lower limit of detection (microCuries per unit mass or volume),

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

E = the counting efficiency (counts per disintegration),

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

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (s^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (s).

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 a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

(2)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~~ plant procedures to assure representative sampling.

Justification for Table 4.11-1 Footnote(2), pg. 3/4 11-3:

Plant procedures for the radioactive release of liquids have been written. These procedures include the methods for mixing to assure representative sampling. The mixing methodology is not in the ODCM.

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)(1) (μCi/ml)
1. Waste Gas Decay Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(2)	1×10^{-4}
2. Containment Purge or Vent	P Each PURGE(3) Grab Sample	P Each PURGE(3)	Principal Gamma Emitters(2)	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
3. Unit Vent	M(3), (4), (5) Grab Sample	M(3)	Principal Gamma Emitters(2)	1×10^{-4}
		M(4) (5)	H-3 (oxide)	1×10^{-6}
~~4. Spent Fuel Building Vent~~	~~M(5) Grab Sample~~	~~M~~	~~Principal Gamma Emitters(2)~~	~~1×10^{-4}~~
		~~M(5)~~	~~H-3 (oxide)~~	~~1×10^{-6}~~
4. ~~5.~~ Radwaste Building Vent	M Grab Sample	M	Principal Gamma Emitters(2)	1×10^{-4}
5. ~~6.~~ All Release Types as listed in 1., 2., 3., 4., and 5. above	Continuous(6)~~(8)~~	W(7) Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous(6)~~(8)~~	W(7) Particulate Sample	Principal Gamma Emitters(2)	1×10^{-11}
	Continuous(6)~~(8)~~ Composite	M Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(6)~~(8)~~	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}

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TABLE 4.11-2 (Continued)

TABLE NOTATIONS (Continued)

(2)The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141, and Ce-144 in iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7, in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(3)Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within 1 hour period.

(4)Tritium grab samples shall be taken and analyzed at least once per 24 hours when the refueling canal is flooded.

(5)Tritium grab samples shall be taken and analyzed at least once per 7 days from the ventilation exhaust from the spent fuel pool area, wherever spent fuel is in the spent fuel pool. Grab samples need to be taken only when spent fuel is in the spent fuel pool.

(6)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.

(7)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. For unit vent, 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% of RATED THERMAL POWER within a 1-hour period, and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3, and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

(8)Continous sampling of the spent fuel building exhaust needs to be performed only when spent fuel is in the spent fuel pool.

Justification for Table 4.11-2

The Spent fuel Building vent is not an effluent release point. The Spent Fuel Building vent exhausts into the Auxiliary Building Ventilation System which exhausts in turn into the Plant Unit vent. The site release point therefore is the Unit vent. The Unit vent is therefore the point where the methodology of the ODCM is applied to assure that the dose limits for Technical Specification 3.11.2.1 are met. Since Table 4.11-2 is designed to aid meeting the criteria of Technical Specification 3.11.2.1, the Spent Fuel Building vent should be deleted. The tritium requirement note for the Spent Fuel vent can therefore be moved to the unit vent.

RADIOACTIVE EFFLUENTS

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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 2 x 10^5 Curies of noble gases (considered as Xe-133 equivalent).
3.3

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, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

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 7 days when radioactive materials are being added and within 7 days following any addition of radioactive material to the tank.

Justification for specification 3.11.2.6, pg. 3/4 11-16:

The meteorological conditions for Wolf Creek allow this value to be increased while still meeting boundary requirements.

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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 whole 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 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations should be made including direct radiation contributions from the 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, 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 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.

4.11.4.2 Cumulative dose contributions from direct radiation from the units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

Justification for specification 3.11.4, pg. 3/4 11-18:

This page was missing.

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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, 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, 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 Specifications 3.11.1.2, 3.11.2.2, or 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 from all radionuclides is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 or 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 required by Specification 6.9.1.6.

c. ~~With the availability of milk or fresh leafy vegetable samples from one or more of the sample locations required by Table 3.12-1 not practicable or possible, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM.~~

Replace with Insert

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

~~The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.~~

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 specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

Insert for specification 3.12.1, Action c., pg. 3/4 12-1,2:

With milk or fresh leafy vegetable samples temporarily unavailable from a routine sampling location, a sample from an alternative location (identified in the ODCM) will be substituted, noting the reason for the unavailability in the Annual Radiological Environmental Operating Report. When changes in sampling locations are permanent, the sampling schedule in the ODCM will be updated to reflect the new routine and alternative sampling locations, and this revision will be described in the Annual Radiological Environmental Operating Report.

Justification for Specification 3.12.1 Action (c), pg. 3/4 12-1,2:

The change in Action (c) is requested to make it clear that temporary sample unavailability at a location (which preoperational experience has shown to occur) does not require revision of the program and the associated paperwork which would occur for actually deleting and/or adding a sampling location.

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne			
Radioiodine and Particulates	Samples from five locations Three samples from close to the three SITE BOUNDARY locations, in different sectors, of the highest calculated annual average ground level D/Q. One sample from the vicinity of a community having the highest calculated annual average ground-level D/Q. One sample from a control location, as for example 15 to 30 km (10 to 20 mile) distant and in the least prevalent wind direction.(3)	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;(4) and gamma isotopic analysis(5) of composite (by location) quarterly.
3. Waterborne			
a. Surface ~~(6)~~ e	One sample upstream (6) and sample downstream.	monthly grab sample ~~Grab sample over 1 month period(7)~~	Gamma isotopic analysis(5) monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from one or two sources only if likely to be affected.(8)	Quarterly.	Gamma isotopic(5) and tritium analysis quarterly.
c. Drinking	One sample of each of one to three of the nearest water supplies that could be affected by its discharge. One sample from a control location.	Composite sample over 2-week period(7) when I-131 analysis is performed, monthly composite otherwise.	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.(9) Composite for gross beta and gamma isotopic analyses(5) monthly. Composite for tritium analysis quarterly.

Justification for Table 3.12-1, 3.a.; pg. 3/4 12-4:

The change in wording and removal of note "7" is requested because the volume of the downstream sampling point, Wolf Creek cooling lake (3.6 x E +10 gallons), is large enough that concentrations of released effluents would fluctuate very gradually with time, making short term or composite sampling unnecessary for this location.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

(5)Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(6)The "upstream sample" shall be taken at a distance beyond significant influenc of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone.

(7)A composite sample is one in which the quantity (aliquot) of liquid sampled is constant over the sampling period and in which the method of sampling employed results in a specimen that is representative of the Average liquid concen-tration ~~trate~~. In this program composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

(8)Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

(9)The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

(10)If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.

Justification for Table 3.12-1, Table notation (7), pg. 3/4 12-8:

Change is requested to correct a typographical error and make the wording more accurate.

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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 that has been approved by the~~ Commission, ~~that~~ correspond to samples required by Table 3.12-1.

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 pursuant to Specification 6.9.1.6.

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

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

Justification for Specification 3/4.12.3, pg. 3/4 12-14:

This page was missing from the final draft.

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HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR, completely offset any rod bow penalties. This margin includes the following:

1) Design limit DNBR of 1.30 vs. 1.28,
2) Grid spacing (K_s) of 0.046 vs. 0.059,
3) Thermal Diffusion Coefficent of 0.038 vs. 0.059,
4) DNBR Multiplier of 0.86 vs. 0.88, and
5) Pitch Reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

FINAL DRAFT

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.9 was determined from expected power control manuevers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^{N}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-3. Measurement errors of 2.1% for RCS total flow rate and 4% for $F_{\Delta H}^{N}$ have been allowed for in determination of the design DNBR value.

DELETE

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venture which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, an inspection is performed of the feedwater venture each refueling outage.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective ACTION is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such ACTION does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

Justification for specification B 3/4.2.2 and B 3/4.2.3

Specification 4.2.3.6 has been requested to be deleted.

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3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit shutdown and maintenance of HOT ~~SHUTDOWN~~ STANDBY of the facility from locations outside of the control room and that a fire will not preclude achieving safe shutdown. The Remote Shutdown System transfer switches, power circuits, and control circuits are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 3 and 19 and Appendix R of 10 CFR Part 50.

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, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection System ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Revision 1, January 1977.

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3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that both adequate warning capability is available for the prompt detection of fires and that Fire Suppression Systems, that are actuated by fire detectors, will discharge extinguishing agents in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

The loss of detection capability for Fire Suppression Systems, actuated by fire detectors represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.9 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the Reactor Coolant System and avoid or mitigate damage to Reactor Coolant System components. The allowable out-of-service times and Surveillance Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 ~~o~~ of Appendix A to 10 CFR Part 50.

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3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be adjusted to values calculated in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitor used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1×10^{-6} µCi/cc are measurable.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Although the orientation of the turbine is such that the number of potentially damaging missiles which could impact and damage safety-related components, equipment, or structures is minimal, protection from excessive turbine overspeed is required.

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PRESSURE/TEMPERATURE LIMITS (Continued)

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = ~~as~~ a function of temperature relative to the RT_{NDT} of the material as provided by the Code,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed

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3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

Insert

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

Insert for specification 3/4.6.1.2, pg. B 3/4 6-1:

For reduced pressure tests, the leakage characteristics yielded by measurements L_{tm} and L_{am} shall establish the maximum allowable test leakage rate L_t of not more than L_a (L_{tm}/L_{am}). In the event L_{tm}/L_{am} is greater than 0.7, L_t shall be specified as equal to L_a $(P_t/P_a)^{1/2}$.

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3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.0 psig, and (2) the containment peak pressure does not exceed the design pressure of 60 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 48 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 49.5 psig, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained in accordance with safety analysis requirements for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 50 psig in the event of a steam line break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of ~~proposed~~ Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979, and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerance on cracking, the results of the engineering evaluation and the corrective actions taken.

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3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 36-inch containment purge supply and exhaust isolation valves are required to be closed and blank flanged during plant operations since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed and blank flanged during plant operation ensures that excessive quantities of radioactive material will not be released via the Containment Purge System. To provide assurance that the 36-inch containment valves cannot be inadvertently opened, the valves are blank flanged.

The use of the containment mini-purge lines is restricted to the 18-inch purge supply and exhaust isolation valves since, unlike the 36-inch valves, the 18-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be
2000 exceeded in the event of an accident during containment purging operation. Operation will be limited to 500 hours during a calendar year. The total time the Containment Purge (vent) System isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons, e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, ~~may~~ should be used to support the additional time requests. Only safety-related reasons should be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4, in any calendar year regardless of the allowable hours.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2.b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA or steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the

FINAL DRAFT

SNUBBERS (Continued)

inspections are augmented by functional testing program, the visual inspection need not be a hands on inspection, but shall require visual scrutiny sufficient to assure that fasteners or mountings for connecting the snubbers to supports or foundations shall have no visible bolts, pins or fasteners missing, or other visible signs of physical damage such as cracking or loosening.

To provide assurance of snubber functional reliability, one of three functional testing methods are used with the stated acceptance criteria:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Put in Insert

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established 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.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Insert for Bases 3/4.7.8:

For mechanical snubbers the force required to initiate or maintain motion of the snubber is not great enough to over-stress the attached piping or component during thermal movement, or to indicate impending failure of the snubber.

Justification for Bases 3/4.7.8, pg. 3/4 7-6:

The insert was added to the bases to clarify that drag force below a specified range is not required to determine a snubbers ability to reduce the effects of seismic events. The snubbers need only allow the component pipe to move without overstressing the component/pipe.

A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

0.015

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and ~~0.05~~ below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

FINAL DRAFT

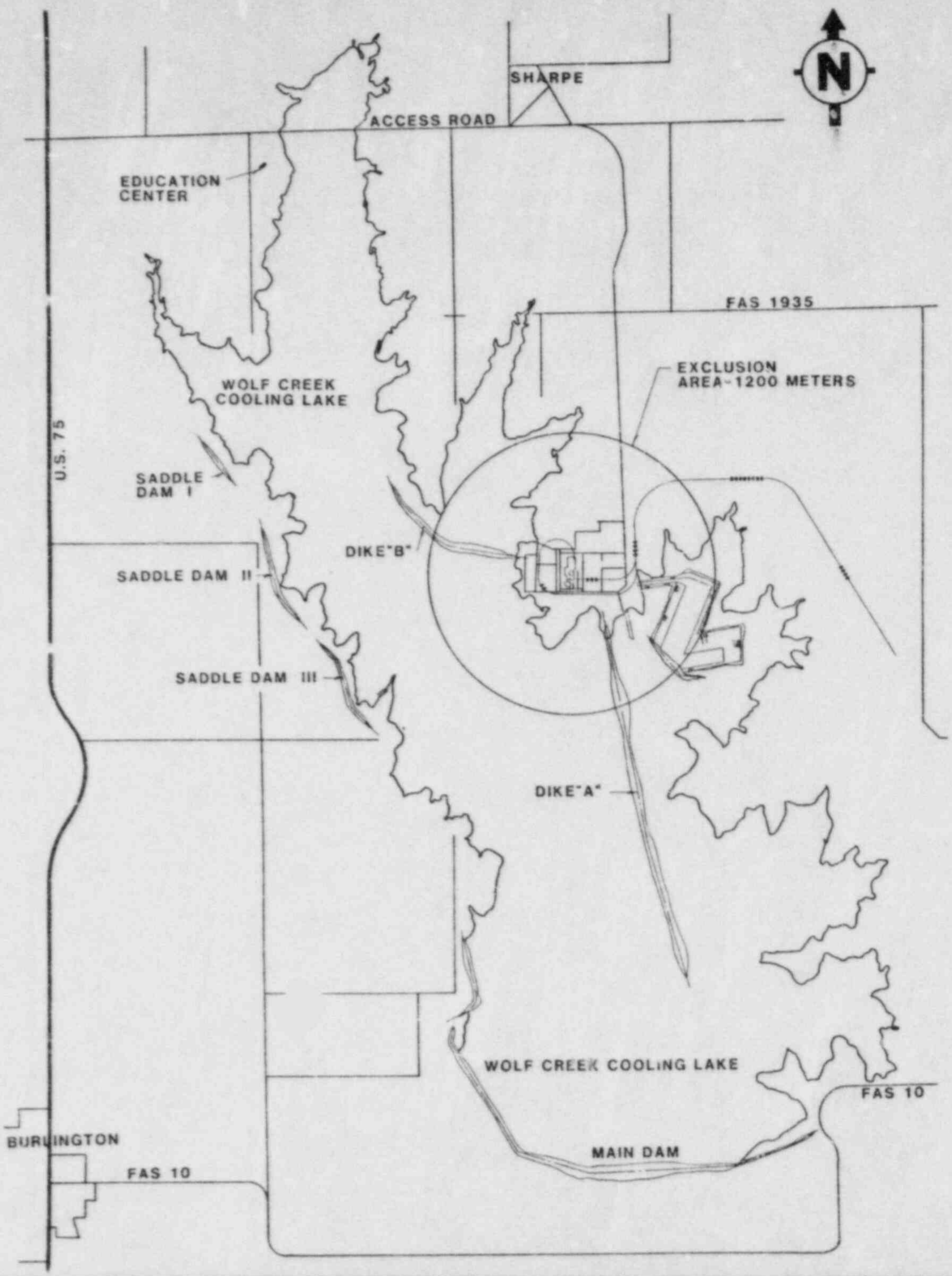
DOSE RATE (Continued)

assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to ~~300~~ 3000 mrems/year to the skin. These release rate limits also restrict, at all times the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in ~~liquid~~ gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive materials in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled



Note:

1. The exclusion-restricted area is a 1200 meter radius circle centered around Unit 1 containment.

FIGURE 5.1-1

EXCLUSION AREA

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at $\leq$ 100°F/h and 200 cooldown cycles at $\leq$ 100°F/h.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 pressurizer cooldown cycles at $\leq$ 200°F/h.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 auxiliary spray acutation cycles.	Spray water temperature differential > 320°F.
	50 leak tests.	Pressurized to $\geq$ 2485 psig.
	5 hydrostatic pressure tests.	Pressurized to $\geq$ 3106 psig.
Secondary Coolant System	1 large steam line break.	Break in a > 6-inch steam line.
	5 hydrostatic pressure tests.	Pressurized to $\geq$ 1350 psig.

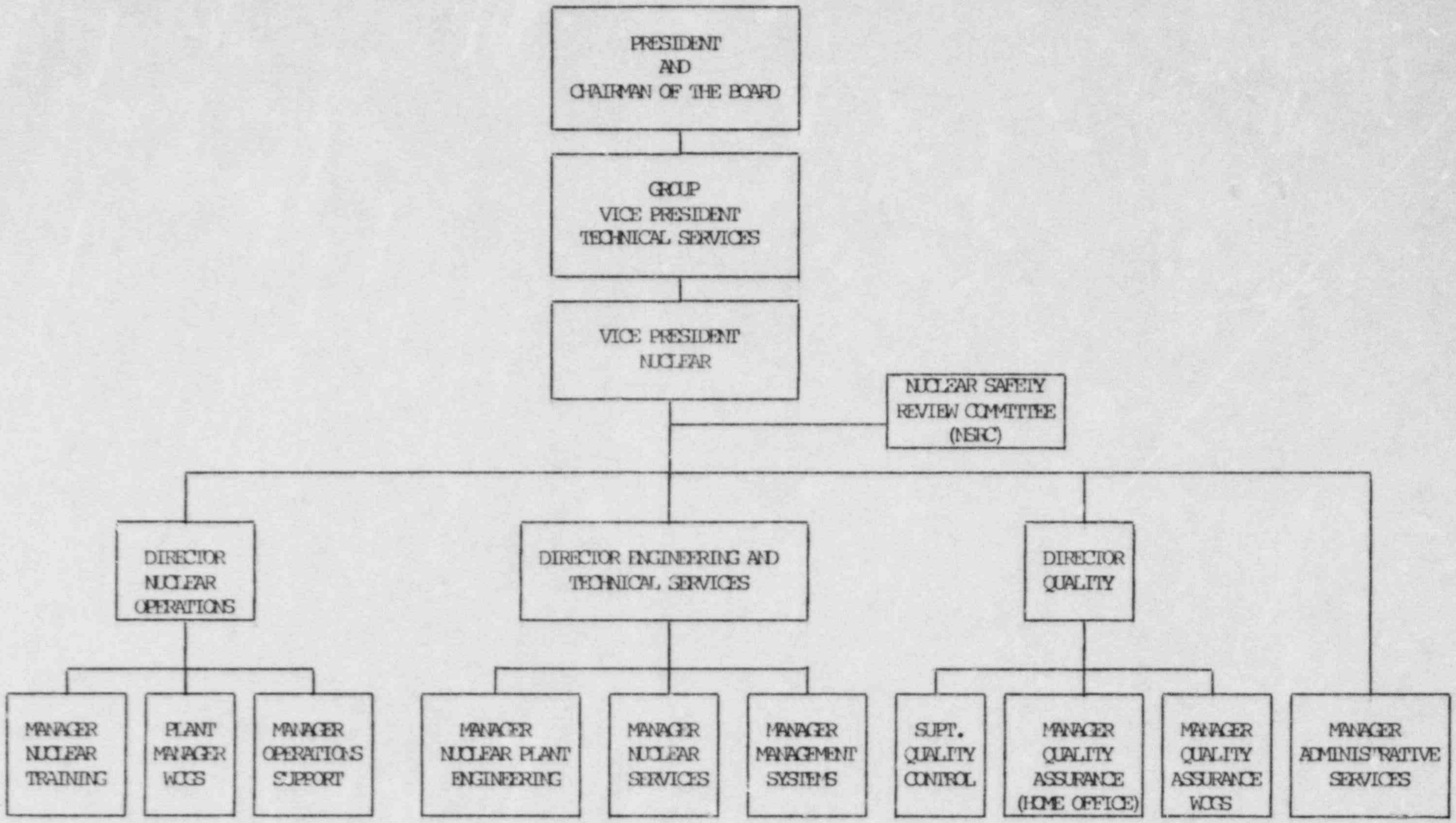


FIGURE 6.2-1

OFFSITE ORGANIZATION

- Plant Manager
  - Supt of Maint
    - Maint Svcs Supvr
      - Maint Engr
      - Mat Cont Supvr
        - Mat Coord Supvr
        - Storeroom Supvr
          - Warehouse Att
      - Utility Supvr
        - Utility Mech
        - Bldg Svcman
    - Maint Supp Supvr
      - Mech Supvr
        - Welder
        - Machinist
        - Mechanic
        - Utility Hlpr
      - Electrical Supv
        - Electrician
        - Utility Hlpr
  - Supt of Operations
    - Ops Coord - Ops#
      - Shift Supervisor
        - Supervising Opr
        - Reactor Operator
        - Station Operator
        - Utility Helper
    - Ops Coord - Planning and Projects
      - Engineer/Spec
      - Surv Coord
        - System Analyst
  - Supt of Tech Supp
    - Reactor Eng Supvr
      - Reactor Engineer
    - I&C Supervisor
      - Engineer/Spec
      - I&C Coordinator
        - I&C Specialist
        - I&C Technician
        - Utility Helper
      - Lead Computer Eng
        - Comp Eng/Spec
        - App Programmer
    - Site Chemist
      - Chemistry Supvr
        - HP/Chem Tech
        - Utility Helper
    - Site Health Phys*
      - HP Supervisor
        - HP/Chem Tech
        - Utility Helper
  - Supt of Plant Supp
    - Fire Prot Spec*
      - Training Spec
    - Chief of Security*
      - Security Ops Supv
      - Lieutenant
        - Sergeant
        - Officer
    - Data Mgmt Supvr
      - Data Base Analyst
      - Asst Supv Comp Ops
        - Technician
        - Computer Opr
      - Asst Supv Sys Sftwr
      - Data Comm Engr Spec
        - Comp Sys Tech
        - Clerk
    - Results Eng Supvr
      - Results Eng (M)
      - Results Eng (E)
  - Supt of Regulatory, Quality and Admin Services
    - Engineering Spec
    - Safety Specialist
    - Nuclear Medical Spec
    - Site Emerg Planning Administrator
    - Document Cont Supvr
      - Document Cont Clerk
      - Analysis & Film Opr
    - Administrative Supvr
      - Clerk
  - Startup Manager
    - Startup Organization

*For technical matters of an immediate nature the respective individual reports directly to the Plant Manager.

#This position requires an SRO License.

FIGURE 6.2-2

UNIT ORGANIZATION

FINAL DRAFT

HIGH RADIATION AREA (Continued)

Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area, or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them, or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Site Health Physicist in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor/Operating Supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed-circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.