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FOR

BABCOCK AND WILCOX

ATMOSPHERIC TYPE CONTAINMENT

REACTOR BUILDING
3/4.6 ~~CONTAINMENT~~ SYSTEMS

REACTOR BUILDING
3/4.6.1 ~~PRIMARY CONTAINMENT~~

REACTOR BUILDING
~~CONTAINMENT~~ INTEGRITY

LIMITING CONDITION FOR OPERATION

REACTOR BUILDING
3.6.1.1 ~~Primary CONTAINMENT~~ INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

REACTOR BUILDING REACTOR BUILDING
Without ~~primary CONTAINMENT~~ INTEGRITY, restore ~~CONTAINMENT~~ INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

6 REACTOR BUILDING
4.~~5~~.1.1 ~~Primary CONTAINMENT~~ INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

1. All penetrations* not capable of being closed by OPERABLE reactor building ~~containment~~ automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification {3.6.4.1}.

~~2. All equipment hatches are closed and sealed.~~

b. By verifying that each reactor building ~~containment~~ air lock is OPERABLE per Specification {3.6.1.3}.

c. The equipment hatch is verified closed and sealed after each opening or at least once per 24 months

*Except valves, blind flanges, and deactivated automatic valves which are located inside the reactor building ~~containment~~ and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that verification of these penetrations being closed need not be performed more often than once per 92 days.

REACTOR BUILDING ~~CONTAINMENT~~ SYSTEMS

REACTOR BUILDING ~~CONTAINMENT~~ LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Reactor building ~~Containment~~ leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. $\leq L_a$, {0.20} percent by weight of the reactor building ~~containment~~ air per 24 hours at P_a, ~~(50)~~ 59 psig, or

2. $\leq L_t$, {0.10} percent by weight of the reactor building ~~containment~~ air per 24 hours at a reduced pressure of P_t, ~~(25)~~ 30 psig.

b. A combined leakage rate of $< 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated ~~containment~~ leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The reactor building ~~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 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 either P_a, 59 ~~(50)~~ psig, or at P_t, 30 ~~(25)~~ psig, during * shall be conducted during the shutdown for the 10-year plant inservice inspection.

* each 10-year service period. The third test of each set

SURVEILLANCE REQUIREMENTS (Continued)

~~b. If any periodic Type A test fails to meet either .75 L_a or .75 L_t, the test schedule for subsequent Type A test shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 L_a or .75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 L_a or .75 L_t at which time the above test schedule may be resumed.~~

b. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 0.25 L_a or 0.25 L_t.

2. Has a duration sufficient to establish accurately the change in leakage between the Type A test and the supplemental test.

3. Required the quantity of gas injected into the ~~containment~~ reactor building or bled from the ~~containment~~ reactor building during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at P_a, ~~(50)~~ 59 psig, or P_t, ~~(25)~~ 30 psig.

c ~~d~~. Type B and C tests shall be conducted with gas at P_a ~~(50)~~ 59 psig at intervals no greater than 24 months except for tests involving air locks.

~~1. Air locks.~~

~~2. Penetrations using continuous leakage monitoring systems.~~

~~3. Valves pressurized with fluid from a seal system.~~

d ~~e~~. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement {4.6.1.3}.

~~f. Type B periodic tests are not required for penetrations continuously monitored by the Containment Isolation Valve and Channel Weld Pressurization Systems, provided the systems are OPERABLE per Surveillance Requirement (4.6.1.4).~~

SURVEILLANCE REQUIREMENTS (Continued)

~~g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, (55) psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.~~

~~h. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a, (50) psig, at intervals no greater than once per 3 years.~~

e ~~i~~. All test leakage rates shall be calculated using observed data converted to absolute values. Error analyses shall be performed to select a balanced integrated leakage measurement system.

f. ~~f~~ ~~j~~. The provisions of Specification 4.0.2 are not applicable.

REACTOR BUILDING ~~CONTAINMENT~~ AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each reactor building ~~containment~~ air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the reactor building ~~containment~~, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a, (59 ~~54~~ psig).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one reactor building ~~containment~~ air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next six hours and in COLD SHUTDOWN within the following 30 hours.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the reactor building ~~containment~~ air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next six hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each reactor building ~~containment~~ air lock shall be demonstrated OPERABLE:

REACTOR BUILDING
~~CONTAINMENT~~ SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

a. *After each opening, except when the airlock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage no greater than 100 ~~175~~ cc/minute when the volume between the door seals is pressurized to $\geq P_a$ (~~54~~ 59 psig) and the rate is determined by either pressure decay for at least 15 minutes or by precision flow measurement when measured for at least 30 seconds with the volume between the door seals at a constant pressure of ~~54~~ 59 psig,

b. At least once per 6 months by conducting an overall air lock leakage test at P_a (~~54~~ 59 psig) and by verifying that the overall air lock leakage rate is within its limit, and

~~c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.~~

*Exemption to Appendix "J" of 10 CFR 50.

~~CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEMS (OPTIONAL)~~

LIMITING CONDITION FOR OPERATIC

3.6.1.4 The containment isolation valve and channel weld pressurization systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment isolation valve or channel weld pressurization system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The containment isolation valve pressurization system shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to $\geq 1.10\ P_a$, (55) psig, and has adequate capacity to maintain system pressure for at least 30 days.

4.6.1.4.2 The containment channel weld pressurization system shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to $\geq P_a$, (50) psig, and has adequate capacity to maintain system pressure for at least 30 days.

REACTOR BUILDING
~~CONTAINMENT~~ SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 The reactor building ~~Primary containment~~ internal pressure shall be maintained between ~~__ and __ psig.~~ 12.0 psia (5.5 in. Hg) and 17.7 psia (3.0 psig).

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

ACTION:

With the reactor building ~~containment~~ internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The reactor building ~~primary containment~~ internal pressure shall be determined to within the limits at least once per 12 hours.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.6 Primary containment average air temperature shall not exceed ___ °F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature > ___ °F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location

a. ______________________

b. ______________________

c. ______________________

d. ______________________

e. ______________________

REACTOR BUILDING ~~CONTAINMENT~~ SYSTEMS

REACTOR BUILDING ~~CONTAINMENT~~ STRUCTURAL INTEGRITY ~~(Typical Dome)~~

LIMITING CONDITIONS FOR OPERATION

3.6.1.7 The structural integrity of the reactor building ~~containment~~ shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the reactor building ~~containment~~ not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Reactor Building ~~Containment~~ Tendons The reactor building ~~containment~~ tendons' structural integrity shall be demonstrated at the end of one, three and five years following the initial containment structural integrity test and at five year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining the lift-off force of ~~that~~ a representative sample of at least 21 tendons (6 dome, 5 vertical, and 10 hoop) ~~each have a lift off force of between ___ (minimum) and ___ (maximum) pounds~~. This test shall include an unloading cycle in which each of these tendons is detensioned to determine if any wires or strands are broken or damaged. If the lift off force of any one tendon in the total sample population is out of the predicted bounds ~~(less than minimum or greater than maximum)~~, an adjacent tendon on each side of the defective tendon shall also be checked for lift off force. If both of these tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. More than one defective tendon out of the original sample population is evidence of abnormal degradation of the reactor building ~~containment~~ structure. Unless there is evidence of abnormal degradation of the reactor building ~~containment~~ structure during the first three tests of the tendons, the number of tendons checked for lift off force during subsequent tests may be reduced to a representative sample of at least 9 tendons (3 dome, 3 vertical and 3 hoop).

REACTOR BUILDING
~~CONTAINMENT~~ SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. Removing one wire or strand from each of the dome, vertical and hoop tendons checked for lift off force and determining that over the entire length of the removed wire or strand:

   1. The tendon wires or strands are free of corrosion.

   2. There are no changes in physical appearance of the sheathing filler grease.

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

4.6.1.7.2 End Anchorages and Adjacent Concrete Surfaces The structural integrity of the end anchorages and adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage concrete exterior surfaces or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A ~~containment~~ leakage rate tests (reference Specification 4.6.1.2) while the ~~containment~~ reactor building is at its maximum test pressure.

4.6.1.7.3 Liner Plate The structural integrity of the ~~containment~~ reactor building liner plate shall be determined during the shutdown for each Type A ~~containment~~ leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the plate and verifying no apparent changes in appearance or other abnormal degradation.

4.6.1.7.4 Reports Any abnormal degradation of the ~~containment~~ reactor building structure detected during the above required tests and inspections shall be reported to the Commission pursuant to Specification ~~[illegible]~~ 6.12.3. This report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, and the corrective actions taken.

A quantitative analytical report covering the results of each the above required inspections shall be submitted to the Commission pursuant to Specification 6.12.4 and shall address, if applicable, broken wires, the force-time trend line for any tendon, when extrapolated, that extends beyond either the upper or lower bounds of the predicted design band, and unexpected changes in tendon conditions or sheathing filler properties.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

REACTOR BUILDING ~~CONTAINMENT~~ SPRAY SYSTEM – OPERATING

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent reactor building ~~containment~~ spray systems shall be OPERABLE with each spray system capable of taking suction from the BWST on an ES ~~containment spray~~ actuation signal and ~~automatically~~ capable of transferring suction to the reactor building ~~containment~~ sump. ~~on a borated water storage tank low level signal. Each spray system flow path from the containment emergency sump shall be via an OPERABLE decay heat cooler.~~

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

ACTION:

With one reactor building ~~containment~~ spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each reactor building ~~containment~~ spray system shall be demonstrated OPERABLE:

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

b. By verifying the operability of ~~, that on recirculation flow,~~ each pump ~~develops a discharge pressure of ≥ ___ psig~~ when tested pursuant to Specification 4.0.5.

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on an ESAS ~~(containment spray test)~~ signal.

2. Verifying the capability of ~~that~~ each spray pump to start~~s~~ automatically on an ESAS ~~(containment spray test)~~ signal.

REACTOR BUILDING
~~CONTAINMENT~~ SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

REACTOR BUILDING
~~CONTAINMENT~~ SPRAY SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.6.2.1 One ~~Two~~ independent reactor building ~~containment~~ spray system~~s~~ shall be OPERABLE with each spray system capable of taking suction from the BWST on an ES ~~containment spray~~ actuation signal and capable of ~~automatically~~ transferring suction to the reactor building ~~containment~~ sump~~, on a borated water storage tank low level signal. Each spray system flow path from the containment emergency sump shall be via an OPERABLE decay heat cooler~~.

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

ACTION:

With no reactor building ~~one containment~~ spray system ~~in~~operable, ~~restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours;~~ restore the inoperable spray system to OPERABLE status within the next ~~48~~ 72 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The reactor building ~~Each containment~~ spray system shall be demonstrated OPERABLE:

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

b. By verifying the operability of the ~~, that on recirculation flow, each~~ pump ~~develops a discharge pressure of ≥ ___ psig~~ when tested pursuant to Specification 4.0.5.

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

1. Verifying that each automatic valve in the flow path actuates to its ~~correct~~ required position on an ESAS ~~(containment spray test)~~ signal.

2. Verifying the capability of the ~~that each~~ spray pump to start~~s~~ automatically on an ESAS ~~(containment spray test)~~ signal.

SURVEILLANCE REQUIREMENTS (Continued)

~~d. At least once per 18 months by verifying a total leak rate ≤ (6) gallons per hour for the system at:~~

~~1. Normal operating pressure of ≥ (350) psig for those parts of the system downstream of the pump suction isolation valve.~~

~~2. ≥ (55) psig for the piping from the containment emergency sump isolation valve to the pump suction isolation valve.~~

d ~~e~~. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

REACTOR BUILDING ~~CONTAINMENT~~ SYSTEMS

SODIUM HYDROXIDE ~~SPRAY~~ ADDITIVE SYSTEM ~~(OPTIONAL)~~

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with a sodium hydroxide ~~spray additive~~ tank~~s~~ ~~containing at least:~~ having an indicated level of $34^{+1.0}_{-0.8}$ ft. of $18^{+3.8}_{-3.0}$ weight per cent solution of sodium hydroxide.

~~a. A contained volume of between (11,300) and ( ) gallons of solution containing between (188,300) and ( ) ppm of sodium hydroxide (NaOH).~~

~~b. A contained volume of between (12,500) and ( ) gallons of solution containing between (287,000) and ( ) ppm of sodium thiosulfate ($Na_2S_2O_3$), between (1,645) and ( ) ppm of boron, and between (5,700) and ( ) ppm of sodium hydroxide (NaOH).~~

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

ACTION:

With the sodium hydroxide ~~spray~~ additive system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the sodium hydroxide ~~spray~~ additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The sodium hydroxide ~~spray~~ additive system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve ~~(manual, power operated or automatic)~~ in the flow path that is not locked, sealed or otherwise secured in position, is in its required ~~correct~~ position. not capable of automatically achieving its required posit

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank~~s~~.

2. Verifying the concentration of the NaOH ~~and $Na_2S_2O_3$~~ solution~~s~~ by chemical analysis.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months, during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on an ~~containment spray test~~ ESAS signal.

~~d. At least once per 5 years by verifying each solution flow rate (to be determined during pre-operational tests) from the following drain connections in the spray additive system:~~

~~1. (Drain line location) ______ ± ______ gpm.~~

~~2. (Drain line location) ______ ± ______ gpm.~~

REACTOR BUILDING
~~CONTAINMENT~~ SYSTEMS

REACTOR BUILDING
~~CONTAINMENT~~ COOLING SYSTEM ~~(OPTIONAL)~~

LIMITING CONDITION FOR OPERATION

3.6.2.3 At least two independent reactor building ~~containment~~ cooling units shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one of the above required reactor building ~~containment~~ cooling units inoperable, restore at least two units to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 At least the above required cooling units shall be demonstrated OPERABLE:

a. At least once per 31 days ~~on a STAGGERED TEST BASIS~~ by:

1. Starting (unless already operating) each unit from the control room.

2. Verifying that each unit operates for at least 15 minutes.

~~3. Verifying a cooling water flow rate of ≥ ___ gpm to each unit cooler.~~

b. At least once per 18 months by verifying that each unit starts automatically ~~(on low sp il~~ upon receipt of an ES ~~test~~ signal.

~~3/4.6.3 IODINE CLEANUP SYSTEM (OPTIONAL)~~

LIMITING CONDITION FOR OPERATION

3.6.3.1 Two independent containment iodine cleanup systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each iodine cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is _____ cfm ±10%.

   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 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying a system flow rate of ____ 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 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < (6) inches Water Gauge while operating the system at a flow rate of ________ cfm ± 10%.

2. Verifying that the system starts on either a safety injection test signal or on a containment pressure - high test signal.

3. Verifying that the filter cooling bypass valves can be opened by operator action.

4. Verifying that the heaters dissipate ________ ± ______ 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 HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of ______ cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of _____ cfm ± 10%.

REACTOR BUILDING
~~CONTAINMENT~~ SYSTEMS

REACTOR BUILDING
3/4.6.4 ~~CONTAINTMENT~~ ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

reactor building
3.6.4.1 The ~~containment~~ isolation valves specified in Table 3.6-1 shall be OPERABLE. ~~with isolation times as shown in Table 3.6-1.~~

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

a. Restore the inoperable valve(s) to OPERABLE status within 48 hours, or

b. Isolate each affected penetration within 48 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 48 hours by use of at least one closed manual valve or blind flange; or

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

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS (Continued)

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

a. Verifying that on an ES ~~containment isolation test~~ signal, each automatic isolation valve actuates to its isolation position.

~~b. Verifying that on a containment radiation high test signal, each purge and exhaust automatic valve actuates to its isolation position.~~

b. Verifying that each manual valve that is not locked, sealed, or otherwise secured in position, is in its required position.

~~4.6.4.1.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.~~

TABLE 3.6-1

REACTOR BUILDING ~~CONTAINMENT~~ ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME
		( ) seconds
A. CONTAINMENT ISOLATION		
1.		
2.		
B. CONTAINMENT PURGE AND EXHAUST		
1.		
2.		
C. MANUAL		
1.		
2.		
D. OTHER		
1.		
2.		

See following pages!

Ref: ANO-1 FSAR

TABLE 5-1

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

#Not subject to Type C leakage tests.

REACTOR BUILDING ISOLATION VALVES

TABLE …6-1

A. REACTOR BUILDING ISOLATION

Penetration No.	Valve No.	Func. (Location to R.B.)
78	CV-1054, 1845	Quench Tank Gas Sample (1054-Inside; 1845-Outside)
9	CV-1270, 1271, 1272, 1273, 1274	RCP Seal Water (1270, 1271, 1272, 1273-Inside) (1274-Outside)
11	CV-4803, 4804	Vent Heater (4803-Inside; 4804-Outside)
14	CV-1214, 1216, 1221	Letdown to Demin (1214, 1216-Inside) (1221-Outside)
25	CV-7453, 7454	Air Particulate Monitor (7453-Inside; 7454-Outside)
39	CV-1065	Quench Tank & Demin. Water Supply (Outside)
40	CV-5611, 5612	Firewater (5611-Outside; 5612-Inside)
41	CV-1667	Nitrogen Supply (Outside)
47	CV-2235	CRD Cooling Water (Outside)
51	CV-6202	Chilled Water to Coolers (Outside)
52	CV-2234	RCP Cooling Water (Outside)
54	CV-2233	Exter. Cooling to Letdown Coolers (Outside)
59	CV-6205, 6203	Chilled Water (6205-Outside, 6203-Inside)
60	CV-2220, 2221	ICW (2220-Outside; 2221-Inside)
62	CV-2214, 2215	ICW (2214-Outside; 2215-Inside)
68	CV-4446, 4440	RB Sump Drain (4446-Inside; 4440-Outside)
70	CV-1053, 1052	Quench Tank Drain (1052-Outside, 1053-Inside)

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A. REACTOR BUILDING ISOLATION (Cont...)

7A	CV-1814*, 1816*	Pzr. & RC Sample (1814 & 1816 - Inside)
10	CV-1820*, 1826*	S.G. Sample (1820, 1826 - Inside)
12	CV-2416*, 2418*	CFT Sample & Bleed (Inside)

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

8. REACTOR BUILDING PURGE AND EXHAUST

Penetration No.	Valve No.	Function	(Location to R.B.)
V-1	CV-7402 CV-7404	Reactor Building Purge " " "	(Outside) (Inside)
V-2	CV-7401 CV-7403	Reactor Building Purge " " "	(Outside) (Inside)
24A	CV-7445 CV-7446	Hydrogen Purge " "	(Outside) (Inside)
24B	CV-7449 CV-7450	Hydrogen Purge " "	(Outside) (Inside)
53A	CV-7443, 7444	Hydrogen Purge	(7443-Outside; 7444-Inside)
53B	CV-7447, 7448	Hydrogen Purge	(7447-Outside; 7448-Inside)

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Penetration No.	Valve No.	Function (Location to R.B.)	
C. MANUAL			
7A	~~CX-1811~~ SS-147	Pzr. & RC Sample	(Outside)
10	SS-146 ~~CX-1822~~	S.G. Sample	(Outside)
12	CF-2	CFT Sample & Bleed	(Outside)
19	SF-42, 43 & 44	Fuel Transfer Canal Recirc. Line	(SF-42 - Outside) (SF-43, -44 - Inside)
31	MU-35A N2-37, -121, -122	CFT "A" Fill CFT "A" N_2 Supply	(Outside) (Inside)
32	MU-35B	CFT "B" Fill ~~CFT "B" N_2 Supply~~	(Outside) ~~(Outside)~~
42	PH-19, PH-20	Plant Heating Return	(PH-19 - Inside; PH-20 - Outside)
43	SA-6, 26	Service Air Supply	(26 - Outside; 6 - Inside)
46	IA-37, 15	Instrument Air Supply	(15 - Inside; 37 - Outside)
48	PH-17, PH-18	Plant Heating Supply	(17 - Outside; 18 - Inside)
58	HV-150, 151	S.G. Blowdown	(150 - Inside; 151 - Outside)
64	HV-139, 146	S.G. Blowdown	(139 - Inside, 146 - Outside)
69	RBD-14, 15, 16, 4, 7A, 9B, 7B, 8C, 9A, 9D, 28	RCS Drain	(RBD-14 - Outside) (RBD-15, 16, 4, 7A, 7B, 9A, 9B, 8C, 9D, 28 ~~Inside~~)

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Penetration No.	Valve No.	Function (Location to R.B.)
D. OTHER (CHECK VALVES)		
3	FW-7B	Main FW to S.G. E24A (Outside)
4	FW-7A	Main FW to S.G. E24B (Outside)
17	FW-13A	Emer. FW to S.G. E24 A (Outside)
31	MU-36A	Core Flood Tank "A" Fill (Inside)
	N2-4	Core Flood Tank "A" N_2 Supply ~~(Inside)~~ (Outside)
32	N2-6	Core Flood Tank "B" Fill ~~(Inside)~~ (Outside)
	MU-36B	Core Flood Tank "B" N_2 Supply (Inside)
39	CS-26	Quench Tank & Demin. Water Supply (Inside)
41	N2-32	Nitrogen Supply (Inside)
47	ICW-30	CRD Cooling Water (Inside)
51	AC-60	Chilled Water to Coolers (Inside)
52	ICW-26	RCP Cooling Water (Inside)
54	ICW-114	Inter. Cooling to Letdown Coolers (Inside)
65	FW-13B	Emer. FW to S.G. E24B (Outside)

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REACTOR BUILDING
~~CONTAINMENT~~ SYSTEMS

3/4.6.5 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.5.1 Two ~~independent containment~~ reactor building hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen analyzer inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 Each hydrogen analyzer shall be demonstrated OPERABLE at least once per ~~92 days on a STAGGERED TEST BASIS~~ 18 months by performing a CHANNEL CALIBRATION using sample gases ~~containing:~~ adequate for calibration over the range of the instrument.

~~a. One volume percent hydrogen, balance nitrogen.~~

~~b. Four volume percent hydrogen, balance nitrogen.~~

~~ELECTRIC HYDROGEN RECOMBINERS - W~~

LIMITING CONDITION FOR OPERATION

3.6.5.2 Two independent containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to $\geq$ 700°F within 90 minutes and is maintained for at least 2 hours.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits.

2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiners (i.e., loose wiring or structural connections, deposits of foreign materials, etc.).

3. Verifying during a recombiner system functional test that the heater sheath temperature increases to $\geq$ 1200°F within 5 hours and is maintained for at least 4 hours.

4. Verifying the integrity of the heater electrical circuits by performing a continuity and resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be $\geq$ 10,000 ohms.

REACTOR BUILDING ~~CONTAINMENT~~ SYSTEMS

HYDROGEN PURGE ~~CLEANUP~~ SYSTEM ~~(If less than 2 hydrogen recombiners available)~~

LIMITING CONDITION FOR OPERATION

3.6.5.3 A reactor building ~~containment~~ hydrogen purge ~~cleanup~~ system shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the reactor building ~~containment~~ hydrogen purge ~~cleanup~~ system inoperable, restore the hydrogen purge ~~cleanup~~ system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3 The hydrogen purge ~~cleanup~~ system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 50 cfm ± 10%

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 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying a system flow rate of 50 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 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <16 ~~(6)~~ inches Water Gauge while operating the system at a flow rate of 50 cfm ± 10%.

~~2. Verifying that the filter cooling bypass valves can be manually opened.~~

3. Verifying that the heaters dissipate 1.0 kw ± 10% ~~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 HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 50 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 50 cfm ± 10%.

~~HYDROGEN MIXING SYSTEM (OPTIONAL)~~

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent hydrogen mixing systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen mixing system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 Each hydrogen mixing system shall be demonstrated OPERABLE:

a. At least once per 92 days on a STAGGERED TEST BASIS by;

   1. Starting each system from the control room.

   2. Verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying a system flow rate of at least ______ cfm.

3/4.6.6 PENETRATION ROOM VENTILATION ~~EXHAUST AIR CLEANUP~~ SYSTEM ~~(OPTIONAL)~~

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent ~~containment~~ penetration room ventilation ~~exhaust air cleanup~~ systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one ~~containment~~ penetration room ventilation ~~exhaust air cleanup~~ system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each ~~containment~~ penetration room ventilation ~~exhaust air cleanup~~ system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by starting each ventilation unit ~~initiating,~~ from the control room, ~~flow through the HEPA filters and charcoal adsorbers~~ and verifying that the system operates for at least 1 ~~10~~ hour~~s with the heaters on.~~

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

~~1. Verifying that with the system operating at a flow rate of ____ cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is ≤ 1% when the system is tested by admitting cold DOP at the system intake. (For systems with diverting valves.)~~

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 2000 cfm ± 10%.

3. 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 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

4. Verifying a system flow rate of 2000 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 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < (6) inches Water Gauge while operating the system at a flow rate of 2000 cfm ± 10%.

2. Verifying that the system starts on an ~~safety injection~~ ES test signal.

3. Verifying that the filter cooling bypass valves can be manually opened.

~~4. Verifying that the heaters dissipate _____ ± _____ 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 HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm $\pm$ 10%.

~~3/4.6.7 VACUUM RELIEF VALVES (OPTIONAL)~~

LIMITING CONDITION FOR OPERATION

3.6.7.1 The primary containment to atmosphere vacuum relief valves shall be OPERABLE with an actuation setpoint of ≤ ___ psid.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one primary containment to atmosphere vacuum relief valve inoperable, restore the valve to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.7.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

Fourteen (14)
3.7.1.1 ~~All~~ main steam line code safety valves shall be OPERABLE. ~~with lift settings as specified in Table 3.7-4.~~

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

less than fourteen (<14)
With ~~one or more~~ main steam line code safety valves ~~in~~operable, ~~operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Nuclear Overpower Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.~~
restore the inoperable valve(s) to OPERABLE status within 24 hours, or place the reactor in at least HOT STANDBY in an additional 12 hours, and in COLD SHUTDOWN within an additional 72 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

~~TABLE 3.7-1~~

~~MAXIMUM ALLOWABLE NUCLEAR OVERPOWER TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES~~

Maximum Number of Inoperable Safety Valves on Any Steam Generator	Maximum Allowable Nuclear Overpower Trip Setpoint (Percent of RATED THERMAL POWER)
1	( )
2	( )
3	( )

~~TABLE 3.7-4~~

~~STEAM LINE SAFETY VALVES PER STEAM GENERATOR~~

VALVE NUMBER	LIFT SETTING (± 1%)*	ORIFICE SIZE
a. ______	______ psig	______
b. ______	______ psig	______
c. ______	______ psig	______
d. ______	______ psig	______

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

EMERGENCY
~~AUXILIARY~~ FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent emergency ~~steam generator auxiliary~~ feedwater pumps and associated flow paths shall be OPERABLE with:

a. One motor-driven emergency ~~auxiliary~~ feedwater pump ~~capable of being powered from an~~ OPERABLE. ~~emergency bus.~~

b. One emergency ~~auxiliary~~ feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

With one emergency ~~auxiliary~~ feedwater system inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each ~~auxiliary~~ feedwater system shall be demonstrated OPERABLE:

a. At least once per 92 ~~31~~ days on a STAGGERED TEST BASIS by:

1. Verifying that the ~~each~~ steam turbine driven pump develops a discharge pressure of $\geq$ 945 psig ~~at a flow of $\geq$ ____ gpm~~ while on recirculation flow when the secondary steam throttle ~~supply~~ pressure is ~~greater than~~ 280 psig ± 10%.

2. Verifying that each valve not capable of automatically achieving its required position ~~(manual, power operated or automatic)~~ in the flow path that is not locked, sealed or otherwise secured in position, is in its ~~correct~~ position.

3. Verifying that the motor-driven emergency feedwater pump develops a discharge pressure of ≥ 945 psig when on recirculation flow.

* The provisions of Specification 3.0.4 may be suspended to allow operational testing of the steam-driven emergency feedwater pump

SURVEILLANCE REQUIREMENTS (Continued)

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on a~~n~~ ~~auxiliary feedwater actuation~~ test signal.

2. Verifying that ~~each~~ the steam-driven pump starts automatically upon receipt of a~~n~~ ~~(auxiliary feedwater actuation~~ test~~)~~ signal.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a minimum contained volume of ~~___ gallons of water.~~ 16.3 feet (107,000 gallons) of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 24 hours either:

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the ~~(alternate source)~~ service water system as a backup supply to the ~~auxiliary~~ emergency feedwater system and restore the condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume to be within its limits when the tank is the supply source for the ~~auxiliary~~ emergency feedwater pumps.

4.7.1.3.2 The ~~(alternate water source)~~ service water system shall be demonstrated OPERABLE at least once per 12 hours by ~~(method dependent upon alternate source) whenever the (alternate water source) is the supply source for the auxiliary feedwater pumps.~~ verifying that at least one service water loop is operating and that the service water system - emergency feedwater system isolation valves are either open or OPERABLE whenever the service water system is the supply source for the emergency feedwater pumps.

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be $\leq$ ~~0.10~~ 0.17 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system > ~~0.10~~ 0.17 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine ~~concentrations greater than 10% of the allowable limit.~~ activity ≥ 5 times background. b) 1 per 6 months, whenever the gross activity determination indicates iodine ~~concentrations below 10% of the allowable limit.~~ activity < 5 times background.

MAIN STEAM BLOCK ~~LINE ISOLATION~~ VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam block ~~line isolation~~ valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

~~MODE 1~~ With one main steam block ~~line isolation~~ valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours. Otherwise, be in HOT SHUTDOWN within the next 12 hours.

~~MODES 2 and 3 With one main steam line isolation valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed provided:~~

~~a. The inoperable isolation valve is maintained closed. Otherwise, be in HOT SHUTDOWN within the next 12 hours.~~

~~b. The provisions of Specification 3.0.4 are not applicable~~.

SURVEILLANCE REQUIREMENTS

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

~~SECONDARY WATER CHEMISTRY~~

LIMITING CONDITION FOR OPERATION

3.7.1.6 The secondary water chemistry shall be maintained within the limits of Table 3.7-2.

APPLICABLITY: MODES 1, 2 and 3.

ACTION:

(To be determined in a manner set forth in the bases and to be imposed by a change to this Specification.)

SURVEILLANCE REQUIREMENTS

4.7.1.6 The secondary water chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.7-3.

~~TABLE 3.7-2~~

~~SECONDARY WATER CHEMISTRY LIMITS~~

Water Sample Location	Parameters*
*	*

~~*Sample locations, parameters and limits to be established in approximatley 6 months following issuance of the full power license based upon test program described in bases.~~

~~TABLE 4.7-3~~

~~SECONDARY WATER CHEMISTRY SURVEILLANCE REQUIREMENTS~~

~~Water sample Location~~	~~Parameters*~~
*	*

~~*Sample locations, parameters and frequencies to be established in approximately 6 months following issuance of the full power license based upon test program described in bases.~~

~~3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION~~

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperature of the secondary coolant in the steam generators shall be > (110)°F when the pressure of the secondary coolant in the steam generator is > (237) psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to ≤ (237) psig within 30 minutes.

b. Perform an engineering evaluation to determine the effect of overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its pressure above (237) psig.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The temperature of the secondary coolant in each steam generator shall be determined to be > (110)°F at least once per hour when secondary pressure in the steam generator is > (237) psig and T_{avg} is < 200°F.

~~3/4.7.3 COMPONENT COOLING WATER SYSTEM~~

LIMITING CONDITION FOR OPERATION

3.7.3.1 Two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one component cooling water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3.1 Each component cooling water loop shall be demonstrated OPERABLE:

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

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on an ESFAS test signal.

2. Verifying that each component cooling water emergency pump starts automatically on an ESFAS test signal.

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 Two independent service water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one service water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

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

b. At least once per 18 months, during shutdown, by performing the Surveillance Requirements of Specification 4.0.5.

~~1. Verifying that each automatic valve in the flow path actuates to its correct position on an ESFAS test signal.~~

~~2. Verifying that each service water emergency pump starts automatically on an ESFAS test signal.~~

EMERGENCY COOLING POND

3/4.7.5 ~~ULTIMATE HEAT SINK (OPTIONAL)~~

LIMITING CONDITION FOR OPERATION

3.7.5.1 The emergency cooling pond ~~ultimate heat sink~~ shall be OPERABLE with:

a. A minimum indicated water ~~level at or above elevation (___) Mean Sea Level, USGS datum.~~ depth of 5 feet (70 acre-feet) with Unit 2 in Modes 1, 2, 3 ... or 3 feet with Unit 2 in Modes 5 or 6.

b. An average water temperature of ≤ 105°F.

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

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The emergency cooling pond ~~ultimate heat sink~~ shall be determined OPERABLE: ~~at least once per 24 hours by verifying the average water temperature and water level to be within their limits.~~

a. At least once per 24 hours by verifying the pond's depth meets the requirements of Specification 3.7.5.1a.

b. At least once per 24 hours during the period of June 1 through September 30 by verifying that the pond's average water temperature is within its limits.

c. At least once per year by making soundings of the pond and verifying the minimum depth meets the requirements of Specification 3.7.5.1a.

~~3/4.7.6 FLOOD PROTECTION (OPTIONAL*)~~

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the _____ (usually the ultimate heat sink) exceeds ____ Mean Sea Level USGS datum, at ____.

APPLICABILITY: At all times.

ACTION:

With the water level at ____ above elevation ____ Mean Sea Level USGS datum:

a. (Be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.)

b. Initiate and complete within ____ hours, the following flood protection measures.

1. (Plant dependent)

2. (Plant dependent)

SURVEILLANCE REQUIREMENTS

4.7.6.1 The water level at ____ shall be determined to be within the limit by:

a. Measurement at least once per 24 hours when the water level is below elevation ___ Mean Sea Level USGS datum, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation ___ Mean Sea Level USGS datum.

* This specification not required if the facility design has adequate passive flood control protection features sufficient to accommodate the Design Basis Flood identified in Regulatory Guide 1.59, August 1973.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CONDITIONING AND AIR FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 Two independent control room emergency air conditioning and air filtration systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one control room emergency air conditioning or air filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7.1.1 Each control room emergency air conditioning system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

   1. Starting each unit from the control room, and

   2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature $\leq$ 84°F D.B.

b. At least once per 18 months by verifying a system flow rate of 9900 cfm $\pm$ 10%.

4.7.7.1.2 Each control room emergency air filtration system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

DRAFT

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ±10%.

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, Revision 2, March 1978.

3. Verifying a system flow rate of 2000 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.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of 2000 cfm ± 10%.

2. Verifying that on a control room high radiation or high chlorine test signal, the system automatically isolates the control room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ± 10%.

~~DRAFT~~

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm $\pm$ 10%.

~~3/4.7.8 ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM~~

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two independent ECCS pump room exhaust cleanup systems trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one ECCS pump room exhaust air cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each ECCS pump room exhaust air cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that with the system operating at a flow rate of ____ cfm $\pm$ 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is $\leq$ 1% when the system is tested by admitting cold DOP at the system intake. (For systems with diverting valves.)

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is ______ cfm ±10%.

3. 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 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

4. Verifying a system flow rate of ____ 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 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < (6) inches Water Gauge while operating the system at a flow rate of ____ cfm ± 10%.

2. Verifying that the system starts on a safety injection test signal.

3. Verifying that the filter cooling bypass valves can be manually opened.

4. Verifying that the heaters dissipate _____ ± _____ 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 HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of ______ cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

f. ~~After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of ______ cfm $\pm$ 10%.~~

3/4.7.9 HYDRAULIC ~~SNUBBERS~~ SHOCK SUPPRESSORS (SNUBBERS)

LIMITING CONDITION FOR OPERATION

3.7.9.1 All hydraulic shock suppressors ~~snubbers~~ listed in Table 3.7-3 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more hydraulic shock suppressors ~~snubbers~~ inoperable, replace or restore the inoperable shock suppressor ~~snubber~~(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.9.1 Hydraulic shock suppressors ~~snubbers~~ will be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Each hydraulic shock suppressors ~~snubber~~ with seal material fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment and approved as such by the NRC, shall be determined OPERABLE ~~at least once after not less than 4 months but within 6 months of initial criticality and~~ in accordance with the inspection schedule of Table 4.7-4 ~~thereafter,~~ by a visual inspection. ~~of the snubber.~~ Visual inspections of the shock suppressors ~~snubbers~~ shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors. ~~Initiation of the Table 4.7-4 inspection schedule shall be made assuming the unit was previously at the 6 month inspection interval.~~

b. Each hydraulic shock suppressor ~~snubber~~ with seal material not fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment shall be determined OPERABLE at least once per 31 days by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown a representative sample of at least 10 hydraulic snubbers or at least 10% of all snubbers listed in Table 3.7-3, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbers greater than 50,000 lbs capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis. Snubbers identified in Table 3.7-3 as either "Especially Difficult to Remove" or in "High Radiation Areas ~~Zones~~" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

see following pages

TABLE 3.7-3

SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS* ~~SNUBBERS*~~

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-3 provided that a revision to Table 3.7-3 is included with the next License Amendment request.

**Modifications to this table due to changes in high radiation areas shall be submitted to the NRC as part of the next License Amendment request.

Table ~~3.16-1~~ 3.7-3

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)*

Snubber No.	Location	Elevation	Snubber in High Radiation Area During Shutdown**	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
HS-1	Decay Heat Line B	329' 1"	X			X
HS-2	Decay Heat Line A	322' 11-3/8"	X			X
HS-49	Decay Heat Line A	329' 1"	X			X
HS-50	Decay Heat Line A	322' 11-3/8"	X			X
HS-8	Pressurizer Spray Line	408' 7-11/16"	X		X	
HS-9	Pressurizer Spray Line	408' 7-11/16"	X		X	
HS-51	Pressurizer Spray Line	373' 0"	X	X	X	
HS-52	Pressurizer Spray Line	373' 0"	X	X	X	
HS-53	Pressurizer Spray Line	382' 0"	X	X	X	
HS-54	Pressurizer Spray Line	381' 6"	X	X	X	
HS-55	Pressurizer Spray Line	398' 6"	X	X	X	
HS-56	Pressurizer Spray Line	398' 0"	X	X	X	
HS-57	Pressurizer Spray Line	406' 10"	X		X	
HS-58	Pressurizer Spray Line	408' 7-11/16"	X		X	
HS-59	Pressurizer Spray Line	408' 7-11/16"	X		X	
HS-60	Pressurizer Spray Line	408' 7-11/16"	X		X	
HS-61	Pressurizer Spray Line	408' 7-11/16"	X		X	
HS-62	Pressurizer Spray Line	408' 7-11/16"	X		X	
HS-63	Pressurizer Spray Line	408' 7-11/16"	X		X	

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-3 provided that a revision to Table 3.7-3 is included with the next License Amendment request.

**Modifications to this table due to changes in high radiation areas shall be submitted to the NRC as part of the next License Amendment request.

Table ~~3.16-1~~ 3.7-3

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)*

bber No.	Location	Elevation	Snubber in High Radiation Area During Shutdown**	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
HS-10	Pressurizer Relief Line	409' 2-3/4"	X		X	
HS-11	Pressurizer Relief Line	410' 2-3/4"			X	
HS-12	Pressurizer Relief Line	410' 2-3/4"			X	
HS-13	Pressurizer Relief Line	400' 0"		X	X	
HS-14	Pressurizer Relief Line	400' 0"		X	X	
HS-66	Pressurizer Relief Line	410' 2-3/4"			X	
HS-67	Pressurizer Relief Line	410' 2-3/4"			X	
HS-68	Pressurizer Relief Line	410' 2-3/4"	X		X	
HS-69	Pressurizer Relief Line	410' 2-3/4"			X	
HS-70	Pressurizer Relief Line	391' 0"	X	X	X	
HS-71	Pressurizer Relief Line	367' 6"	X	X	X	
HS-72	Pressurizer Relief Line	357' 0"	X	X	X	
HS-88	Pressurizer Relief Line	370' 0"	X	X	X	
H-A-1	Pressurizer Relief Line	400' 0"	X	X	X	
H-A-2	Pressurizer Relief Line	399' 0"	X	X	X	
H-B-1	Pressurizer Relief Line	400' 0"	X	X	X	
H-B-2	Pressurizer Relief Line	391' 0"	X	X	X	
H-C-1	Pressurizer Relief Line	410' 2-3/4"		X	X	
H-C-2	Pressurizer Relief Line	394' 0"		X	X	
HS-3	Main Steam Line A	425' 0"		X	X	
HS-4	Main Steam Line A	408' 6"			X	
HS-5	Main Steam Line A	42[illegible]' 0"				X
HS-7	Main Steam Line B	42[illegible]' 0"				X
HS-15	Main Steam Line A	408' 6"			X	
HS-16	Main Steam Line B	423' 2"		X	X	
HS-17	Main Steam Line B	42[illegible]' 2"		X	X	
HS-18	Main Steam Line B	408 6"			X	
HS-19	Main Steam Line B	396 0"		X	X	
HS-20	Main Steam Line B	408' 6"			X	
HS-22	Main Feedwater Header B	376' 4-11/16"		X	X	
HS-23	Main Feedwater Header B	376' 4-11/16"		X	X	

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-3 provided that a revision to Table 3.7-3 is included with the next License Amendment request.

Table ~~3.~~ ~~(Cont.)~~ 3.7-3

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)*

bber No.	Location	Elevation	Snubber in High Radiation Area During Shutdown**	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
S-24	Main Feedwater Header B	376' 4-11/16"	X	X	X	
HS-25	Main Feedwater Header B	376' 4-11/16"	X	X	X	
HS-26	Main Feedwater Header B	376' 4-11/16"		X	X	
S-27	Main Feedwater Header B	376' 4-11/16"		X	X	
HS-28	Main Feedwater Header B	376' 4-11/16"	X	X	X	
S-29	Main Feedwater Header B	376' 4-11/16"	X	X	X	
S-30	Main Feedwater Line A	361' 0"			X	
HS-31	Main Feedwater Header A	376' 4-11/16"		X	X	
HS-32	Main Feedwater Header A	376' 4-11/16"		X	X	
HS-33	Main Feedwater Header A	376' 4-11/16"		X	X	
S-34	Main Feedwater Header A	376' 4-11/16"		X	X	
HS-35	Main Feedwater Header A	376' 4-11/16"		X	X	
HS-36	Main Feedwater Header A	376' 4-11/16"	X	X	X	
HS-37	Main Feedwater Header A	376' 4-11/16"		X	X	
HS-38	Main Feedwater Header A	376' 4-11/16"		X	X	
HS-21	Emergency Feedwater Line B	394' 0"	X	X	X	
1A	Reactor Coolant Pump A	390' 10"		X	X	
2A	Reactor Coolant Pump A	390' 10"		X	X	
1B	Reactor Coolant Pump B	390' 10"		X	X	
2B	Reactor Coolant Pump B	390' 10"		X	X	
1C	Reactor Coolant Pump C	390' 10"		X	X	
2C	Reactor Coolant Pump C	390' 10"		X	X	
1D	Reactor Coolant Pump D	390' 10"		X	X	
2D	Reactor Coolant Pump D	390' 10"		X	X	

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-3 provided that a revision to Table 3.7-3 is included with the next License Amendment request.

**Modifications to this table due to changes in high radiation areas shall be submitted to the NRC as part of the next License Amendment request.

TABLE 4.7-4

SHOCK SUPPRESSOR
HYDRAULIC ~~SNUBBER~~ INSPECTION SCHEDULE

SHOCK SUPPRESSORS NUMBER OF ~~SNUBBERS~~ FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*	NEXT REQUIRED INSPECTION INTERVAL**
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3 or 4	124 days ± 25%
5, 6, or 7	62 days ± 25%
≥8	31 days ± 25%

* Snubbers may be categorized into two groups, "accessible" and "inaccessible". This categorization shall be based upon the snubber's accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

** The required inspection interval shall not be lengthened more than one step at a time and the provisions of Specification 4.0.2 are not applicable.

3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.10.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.10.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.10.1.2 Test Frequencies - Each category of sealed sources shall be tested at the frequency described below.

a. Sources in use (excluding startup sources and fission detectors previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive material:

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3) and

2. In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.10.1.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

3/4.8.1 A.C. SOURCES

~~OPERATING~~

LIMITING CONDITION FOR OPERATION

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

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system.

b. Two separate and independent diesel generators each with:

1. A ~~separate~~ separate day ~~and engine-mounted fuel~~ tank~~s~~ containing a minimum volume of 160 gallons of fuel.

2. A separate emergency fuel storage system ~~containing~~ with a minimum ~~volume~~ level of ~~____ gallons of fuel.~~ 138 inches (20 000 gallons) of fuel.

3. A separate fuel transfer pump.

4. A starting air compressor.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With ~~either~~ an offsite circuit ~~or diesel generator~~ of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining ~~A.C. sources by~~ offsite circuit performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 unless the diesel generator is already operat. within one hour and at least once per ~~8~~ 24 hours thereafter; restore at least two offsite circuits ~~and two diesel generators~~ to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~b. With a diesel generator ... of the remaining diesel generator performing ... restore at least~~

~~c. b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate~~ the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a ~~and~~ 4.8.1.1.2.a.4 unless the diesel generat is already operating. within one hour and at least once ~~per~~ ~~8~~ 24 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in ~~at~~ least HOT STANDBY ~~within~~ the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ~~Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss~~

ACTION: (Continued)

b. With a diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining diesel generator by performing Surveillance Requirements 4.8.1.1.1.a and unless the diesel generator is already operating, 4.8.1.1.2.a.4 within one hour and at least once per 24 hours thereafter; restore at least two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and unless the diesel generator is already operating, 4.8.1.1.2.a.4 within one hour and at once per 24 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ELECTRICAL POWER SYSTEMS

ACTION (Continued)

~~or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

d. With both ~~two~~ of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within one hour and at least once per 24 ~~8~~ hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or reduce power to house load ~~be in at least HOT STANDBY~~ within the next 6 hours. ~~With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

e ~~d~~. With both ~~two~~ of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 24 ~~8~~ hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 6 ~~2~~ hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ~~Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availability.

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a {STAGGERED TEST BASIS} by:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying the fuel volume in the day ~~and engine-mounted fuel~~ tank.

2. Verifying the fuel volume in the ~~fuel~~ emergency storage tank.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day ~~and engine-mounted~~ tank.

4. Verifying the diesel starts from ambient condition and accelerates to at least (900) rpm in ≤ ~~(10)~~ 15 seconds.

5. Verifying the generator is synchronized, loaded to ≥ 1350 ~~(50)%~~ kw, and operates for ≥ 60 minutes.

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

~~7. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block is within ± 10% of its design interval.~~

b. At least once per 92 days by verifying that a sample of diesel fuel obtained ~~in accordance with ASTM-D270-63,~~ from the ~~fuel~~ emergency storage tank is within the acceptable limits specified in Table 1 of ASTM D975-~~74~~ 68 when checked for viscosity, water and sediment.

c. 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 generator capability to reject a load of ≥ 522 ~~≥ (largest single emergency load)~~ kw without tripping.

3. Simulating a loss of offsite power in conjunction with a safety injection actuation test signal, and:

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for ≥ 5 minutes while its generator is loaded with the emergency loads.

SURVEILLANCE REQUIREMENTS (Continued)

c) Verifying that all diesel generator trips, except engine overspeed and generator differential, are automatically bypassed ~~upon loss of voltage~~ on the emergency bus and/or a safety injection actuation test signal.

4. Verifying the diesel generator operates for ≥ 60 minutes while loaded to ~~≥ (100)% kw.~~ full rating.

~~5. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of ___ kw.~~

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.

b. One diesel generator with:

1. Day and engine-mounted fuel tanks containing a minimum volume of ____ gallons of fuel.

2. A fuel storage system containing a minimum volume of ____ gallons of fuel.

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the minimum required A.C. electrical power sources are restored to OPERABLE status.

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 Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2, except for requirement 4.8.1.1.2a.5.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION ~~OPERATING~~

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized with tie breakers open between redundant busses:

{4160}	volt Emergency Bus #	A3
{4160}	volt Emergency Bus #	A4
{480}	volt Emergency Bus #	B5
{480}	volt Emergency Bus #	B6
~~{240}~~	~~volt A.C. Vital Bus #~~	
~~{240}~~	~~volt A.C. Vital Bus #~~	
{120}	volt A.C. Vital Bus #	RS1
{120}	volt A.C. Vital Bus #	RS2
{120}	volt A.C. Vital Bus #	RS3
{120}	volt A.C. Vital Bus #	RS4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 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 The specified A.C. busses shall be determined OPERABLE with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment and indicated power availability.

~~A.C. DISTRIBUTION - SHUTDOWN~~

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE:

1 - (4160) volt Emergency Bus

1 - (480) volt Emergency Bus

1 - (240) volt A. C. Vital Bus

2 - (120) volt A.C. Vital Busses.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

D.C. DISTRIBUTION ~~- OPERATING~~

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. bus trains shall be energized and OPERABLE with tie breakers between bus trains open:

TRAIN "A" consisting of {~~250/~~125}-volt D.C. bus No. 1, {~~250/~~125}-volt D.C. battery bank No. 1 and a full capacity charger.

TRAIN "B" consisting of {~~250/~~125}-volt D.C. bus No. 2, {~~250/~~125}-volt D.C. battery bank No. 2 and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one {~~250/~~125}-volt D.C. bus inoperable, restore the inoperable bus to OPERABLE status within 6 ~~2~~ hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one {~~250/~~125}-volt D.C. battery and/or its charger inoperable, restore the inoperable battery and/or charger to OPERABLE status within 6 ~~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

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized with tie breakers open at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.3.2 Each {~~250/~~125}-volt battery bank and charger shall be demonstrated OPERABLE:

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

1. The electrolyte level of each pilot cell is between the minimum and maximum level indication marks.

SURVEILLANCE REQUIREMENTS (Continued)

2. The pilot cell specific gravity, corrected to {77}°F and full electrolyte level, is ≥ 1.20.

3. The pilot cell voltage is ≥ 2.13 volts.

4. The overall battery voltage is ≥ ~~{250/~~125} volts.

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is ≥ 2.13 volts under float charge. ~~and has not decreased more than ___ volts from the value observed during the original acceptance test~~.

2. The specific gravity, corrected to {77}°F and full electrolyte level, of each connected cell is ≥ 1.20. ~~and has not decreased more than ___ from the value observed during the previous test.~~

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or deterioration.

2. The cell-to-cell and terminal connections are clean, tight and coated with anti-corrosion material.

3. The battery charger will supply at least 169 amperes at a minimum of 105 volts for at least ~~8 hours~~. 30 minutes.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for {8} hours when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test.

~~D.C. DISTRIBUTION - SHUTDOWN~~

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

1 - (250/125)-volt D.C. bus.

1 - (250/125)-volt battery bank and charger supplying the above D.C. bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required (250/125)-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required (250/125)-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration ~~of all filled portions~~ of the r~~R~~eactor c~~C~~oolant ~~System and the refueling canal~~ shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, ~~which includes a 1% $\Delta k/k$ conservative allowance for uncertainties,~~ or

b. A boron concentration of $\geq$ ~~(1850) ppm, which includes a 50 ppm conservative allowance for uncertainties.~~ 1800 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at $\geq$ {25} gpm of {8100} ppm boron or its equivalent until K_{eff} is reduced to $\leq$ 0.95 or the boron concentration is restored to $\geq$ ~~(1850)~~ 1800 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head.

b. Withdrawal of any safety or regulating rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant ~~system~~ and the refueling canal shall be determined by chemical analysis at least once each 72 hours.

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*The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication. ~~in the control room and one with audible indication in the containment and the control room.~~

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per ~~7~~ 14 days.

b. A CHANNEL FUNCTIONAL TEST within ~~8~~ 24 hours prior to the initial start of CORE ALTERATIONS.

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least: ~~(100) hours.~~

a. 72 hours when discharging ≤ 177 irradiated assemblies with < 413 irradiated assemblies in the spent fuel pool, or

b. 175 hours when discharging 177 irradiated assemblies with 413 irradiated assemblies in the spent fuel pool.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than the above required time, ~~(100) hours,~~ suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least the time required by Specification 3.9.3 ~~(100) hours~~ by verification of the date and time of subcriticality and the number of irradiated assemblies in the spent fuel pool prior to movement of irradiated fuel in the reactor pressure vessel.

REACTOR BUILDING ~~CONTAINMENT~~ PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor building ~~containment~~ penetrations shall be in the following status:

a. The equipment hatch ~~door~~ closed and held in place by a minimum of four bolts,

b. A minimum of one door in the personnel and emergency hatches ~~each airlock~~ closed.

~~c. Each penetration providing direct access from the reactor building containment atmosphere to the outside atmosphere shall be either:~~

~~1. Closed by an isolation valve, blind flange, or manual valve, or~~

~~2. Be capable of being closed by an OPERABLE automatic containment purge and exhaust isolation valve.~~

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the ~~containment.~~ reactor building.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the ~~containment.~~ reactor building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required reactor building ~~containment~~ penetrations shall be determined to be either in its closed/isolated condition or capable of being closed ~~by an OPERABLE automatic containment purge and exhaust valve~~ within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment by ~~a.~~

~~a.~~ Verifying the penetrations are in their isolated condition, or

~~b. Testing the containment purge and exhaust valves per the applicable portions of Specification (4.6.4.1.2).~~

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

~~FUEL HANDLING BRIDGE OPERABILITY~~

LIMITING CONDITION FOR OPERATION

3.9.6 The fuel handling bridges shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. A hoist minimum capacity of ____ pounds.

b. A hoist overload cutoff limit of ____ pounds.

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for bridge OPERABILITY not satisfied, suspend use of any inoperable bridge from operations involving the movement of control rods or fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6 Each fuel handling bridge used for movement of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of moving control rods or fuel assemblies by performing a hoist load test of at least ____ pounds and demonstrating an automatic load cutoff when the hoist load exceeds ____ pounds.

AUXILIARY

CRANE TRAVEL - ~~SPENT FUEL STORAGE POOL~~ BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the ~~storage~~ spent fuel pool.

APPLICABILITY: With fuel assemblies and water in the ~~storage~~ spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 The crane electrical power disconnect which prevents crane travel over the spent fuel pool shall be verified open under administrative control at least once per 7 days, or the crane travel interlock which prevents crane travel over the spent fuel pool shall be demonstrated OPERABLE within 4 hours prior to each use of the crane for lifting loads in excess of 2000 pounds.

COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8 At least one decay heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one decay heat removal loop in operation, except as provided in b below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all reactor building ~~containment~~ penetrations providing direct access from the reactor building ~~containment~~ atmosphere to the outside atmosphere within 4 hours.

b. The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS. ~~in the vicinity of the reactor pressure vessel (hot) legs.~~

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 A decay heat removal loop shall be demonstrated to be operating and circulating reactor coolant at a flow rate of ≥ ~~(2800)~~ 1500 gpm at least once per 24 hours.

~~CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM~~

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge and exhaust isolation system shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With the containment purge and exhaust isolation system inoperable, close each of the purge and exhaust penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge and exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge and exhaust isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

~~WATER LEVEL - REACTOR VESSEL~~

LIMITING CONDITION FOR OPERATION

3.9.10 As a minimum, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operation involving movement of fuel assemblies or control rods within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during movement of fuel assemblies or control rods within the reactor pressure vessel.

REFUELING OPERATIONS

~~STORAGE POOL WATER LEVEL~~

LIMITING CONDITION FOR OPERATION

3.9.11 As a minimum, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel and crane operations with loads in the fuel storage area and restore the water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

FUEL HANDLING AREA VENTILATION ~~STORAGE POOL AIR CLEANUP~~ SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The fuel handling area ventilation ~~Two independent fuel storage pool air cleanup~~ ~~systems~~ system shall be OPERABLE.

APPLICABILITY: Whenever handling irradiated fuel ~~is~~ in the storage pool.

ACTION:

~~a. With one fuel storage pool air cleanup system inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE fuel storage pool air cleanup system is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.~~

a. ~~b.~~ With the ~~no~~ fuel handling area ventilation ~~storage pool air cleanup~~ system IN OPERABLE, suspend all operations involving movement of irradiated fuel within the ~~storage~~ spent fuel pool ~~or crane operation with loads over the storage pool~~ until the ~~at least one~~ fuel handling area ventilation ~~storage pool air cleanup~~ system is restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel handling area ventilation ~~storage pool air cleanup~~ ~~systems~~ system shall be demonstrated OPERABLE:

~~a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.~~

a. ~~b.~~ At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of 40 000 cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is ≤ 1% when the system is tested by admitting cold DOP at the system intake. (For systems with diverting valves.)

2. Verifying that the ~~cleanup~~ ventilation system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 40 000 cfm ±10%.

3. 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 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

4. Verifying a system flow rate of 40 000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

b. 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 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

c ~~d~~. At least once per 18 months by ~~:~~

~~1.~~ ~~V~~verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < {6} inches Water Gauge while operating the system at a flow rate of 40,000 cfm ± 10%.

~~2. Verifying that on a high radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.~~

SURVEILLANCE REQUIREMENTS (Continued)

~~3. Verifying that the system maintains the spent fuel storage pool area at a negative pressure of ≥ (1/4) inches Water Gauge relative to the outside atmosphere during system operation.~~

~~4. Verifying that the filter cooling bypass valves can be manually opened.~~

~~5. Verifying that the heaters dissipate ____ ± ____ kw when tested in accordance with ANSI N510-1975.~~

d ~~g~~. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 40 000 cfm ± 10%.

e ~~f~~. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 40 000 cfm ± 10%.

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.1 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.2, 3.1.3.6, 3.1.3.7, 3.1.3.8, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained ≤ 85% of RATED THERMAL POWER.

b. The Nuclear Overpower Trip Setpoint is ≤ 10% of Rated Thermal Power higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% of RATED THERMAL POWER.

~~c. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in 4.10.1.2 below.~~

APPLICABILITY: MODE 1 (during PHYSICS TEST).

ACTION:

With the thermal power > 85% of Rated Thermal Power & ~~any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.2, 3.1.3.6, 3.1.3.7, 3.1.3.8, 3.2.1 or 3.2.4 are suspended, either:~~

a. Reduce THERMAL POWER to ≤ 85% of Rated Thermal Power, or ~~sufficiently to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or~~

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

SURVEILLANCE REQUIREMENTS

4.10.1.1 The Nuclear Overpower Trip Setpoint shall be determined to be set within the limits specified within 8 hours prior to the initiation of ~~and at least once per 8 hours during~~ PHYSICS TESTS.

~~4.10.1.2 The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at least once per two hours during PHYSICS TESTS.~~

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.2 The limitations of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.6, 3.1.3.7, and 3.1.3.8 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

b. The reactor trip setpoints on the OPERABLE Nuclear Overpower Channels are set at ≤ 25% of RATED THERMAL POWER.

c. The nuclear instrumentation Source Range and Intermediate Range high startup rate control rod withdrawal inhibit are OPERABLE.

APPLICABILITY: MODE 2 (during PHYSICS TEST).

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the control rod drive trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 Each Source and Intermediate Range and Nuclear Overpower Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTION ~~(OPTIONAL)~~

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification ~~3.4.1~~ 3.1.1.4 and 3.3.1 may be suspended during the performance of ~~startup and~~ PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, ~~and~~

b. The reactor trip setpoints on the OPERABLE Nuclear Overpower channels are set ≤ 25% of RATED THERMAL POWER.

APPLICABILITY: During ~~startup and~~ PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately ~~open the control rod drive trip breakers.~~ trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be < 5% of RATED THERMAL POWER at least once per hour during ~~startup and~~ PHYSICS TESTS.

4.10.3.2 Each Nuclear Overpower Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating ~~startup or~~ PHYSICS TESTS.

SPECIAL TEST EXCEPTION

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.4 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided:

a. Reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

b. All axial power shaping rods are withdrawn to at least 90% (indicated position) and OPERABLE.

APPLICABILITY: MODE 2.

ACTION:

a. With any safety or regulating control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion or the axial power shaping rods not within their withdrawal limits, immediately initiate and continue boration at $\geq$ {25} gpm of {8700} ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all safety or regulating control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ {25} gpm of {8700} ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The position of each safety, regulating, and axial power shaping rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.4.2 Each safety or regulating control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.4.3 The axial power shaping rods shall be demonstrated OPERABLE by moving each axial power shaping rod > 6.5% (~~indicated position~~ 9.0 inches) within 4 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.5 The minimum temperature for criticality limits of Specification 3.1.1.~~5~~ 4 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Nuclear Overpower ~~Linear Power Level - High neutron flux monitoring~~ channels are set at $\leq$ 20% of RATED THERMAL POWER, and

~~c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation shown on Figure 3.4-2.~~

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

a. With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

~~b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figure 3.4-2, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the engineering evaluation required by Specification 3.4.9.1 prior to the next reactor criticality.~~

SURVEILLANCE REQUIREMENTS

~~4.10.5.1 The Reactor Coolant System temperature and pressure relationship shall be verified to be within the acceptable region for operation of Figure 3.4-2 at least once per hour.~~

4.10.5.1 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour.

4.10.5.2 Each Nuclear Overpower ~~Logarithmic Power Level and Linear Power Level~~ channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

BASES

FOR

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The summary statements contained in this section provide the bases for the specifications of Sections 3.0 and 4.0 and are not considered a part of these technical specifications as provided in 10 CFR 50.36.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification {3.5.1} calls for each Reactor Coolant System core flooding tank to be OPERABLE and provides explicit ACTION requirements when one tank is inoperable. Under the terms of Specification 3.0.3, if more than one tank is inoperable, the facility is required to be in at least HOT STANDBY within 1 hour and in COLD SHUTDOWN within the following 30 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

BASES

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operations. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

BASES

Under the t[illegible]s of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these technical specification.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes predence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. During Modes 1 and 2 the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident a minimum SHUTDOWN MARGIN of ~~(0.30)~~ (LATER) % $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

3/4.1.1.2 BORON DILUTION

A minimum flow rate of at least ~~(2000)~~ 1500 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual through the Reactor Coolant System in the core during boron concentration reductions in the Reactor Coolant System. A flow rate of at least ~~(2000)~~ 1500 GPM will circulate an equivalent Reactor Coolant System volume of ~~(1[illegible],000)~~ 11,000 cubic feet in approximately ~~30~~ 60 minutes. The reactivity change rate associated with boron concentration reduction will be within the capability for operator recognition and control.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.~~1~~.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 525°F. This limitation is required to ensure ~~1)~~ the moderator temperature coefficient is within its analyzed temperature range. ~~2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.~~

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) makeup or DHR pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE emergency busses.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.0 %Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires either 550 ft^3 (73 inches) ~~gallons~~ of 8700 ppm borated water from the boric acid addition ~~storage~~ tanks or 16000 gallons of 2270 ppm borated water from the borated water storage tank.

The requirements for a minimum contained volume of 350,000 ~~(402,500)~~ gallons (35.9 indicated feet) of borated water in the borated water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified.

~~With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the~~

3/4.1.2 BORATION SYSTEMS (Continued)

~~stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.~~

~~The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either ( ) gallons of (12,250) ppm borated water from the boric acid storage system or ( ) gallons of (1800) ppm borated water from the borated water storage tank.~~

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The limits on contained water volume, and boron concentration ensure a pH value of between 9.0 ~~(8.5)~~ and 10.4 ~~(11.0)~~ of the solution sprayed within the reactor building ~~containment~~ after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

~~The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.~~

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. For example, misalignment of a safety or regulating rod requires a restriction in THERMAL POWER. In addition, those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

The maximum rod drop time permitted is consistent with or conservative with respect to the assumed rod drop time used in the safety analyses. Measurement with T_{avg} $\geq$ {525}°F and with reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The limitation on THERMAL POWER based on xenon reactivity is necessary to ensure that power peaking limits are not exceeded even with specified rod insertion limits satisfied.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core ≥ {~~1.32/~~1.30} during normal operation and during short term transients, (b) maintaining the peak linear power density ≤ 18.0 kw/ft during normal operation, and (c) maintaining the peak power density ≤ ~~(23.5)~~ 19.4 kw/ft during short term transients. In addition, the above criteria must be met in order to meet the assumptions used for the loss-of-coolant accidents.

The power-imbalance envelope defined in Figures {3.2-1, ~~and~~ 3.2-2}, and 3.2-3, and the insertion limit curves, Figures {3.1-1, 3.1-2, 3.1-3, 3.1-4, ~~and~~ 3.1-5, and 3.1 ~~3.1-5}~~, are based on LOCA analyses which have defined the maximum linear heat rate such that the maximum clad temperature will not exceed the Final Acceptance Criteria of 2200°F following a LOCA. Operation outside of the power-imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The power-imbalance envelope represents the boundary of operation limited by the Final Acceptance Criteria only if the control rods are at the insertion limits, as defined by Figures {3.1-1, 3.1-2, 3.1-3, 3.1-4, ~~and~~ 3.1-5}, and 3.1-[illegible] and if a ~~(4)~~ 3.4 percent QUADRANT POWER TILT exists, simultaneously with all other engineering and uncertainty factors also at their limits. Additional conservatism is introducted by application of:

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing tolerance factors.

e. Fuel rod bowing.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensures that the original criteria are met.

The definitions of certain design limit nuclear power peaking factors as used in these specifications are as follows:

F_Q Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It has been determined by extensive analysis of possible operating power shapes that the design limits on nuclear power peaking and on minimum DNBR at full power are met, provided:

$F_Q \leq \cancel{\{2.50\}}\ 2.67$; $F^N_{\Delta H} \leq \cancel{\{1.71\}}\ 1.78$ $F^N_z = 1.50$

Power Peaking is not a directly observable quantity and therefore limits have been established on the bases of the AXIAL POWER IMBALANCE produced by the power peaking. It has been determined that the above hot channel factor limits will be met provided the following conditions are maintained.

1. Control rods in a single group move together with no individual rod insertion differing by more than ~~± ( )% (indicated position)~~ ±9.0 inches from the group average height.

2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.

3. The regulating rod insertion limits of Specification 3.1.3.6 are maintained.

4. AXIAL POWER IMBALANCE limits are maintained. The AXIAL POWER IMBALANCE is a measure of the difference in power between the top and bottom halves of the core. Calculations of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with AXIAL POWER IMBALANCE. The correlation shows that the design power shape is not exceeded if the AXIAL POWER IMBALANCF is maintained ~~between + (10) percent and - (15) percent at RATED THERMAL POWER.~~ within the limits of Figures 3.2-1, 3.2-2, and 3.2

~~The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod insertion and are the core DNBR design basis. Therefore, for operation at a fraction of RATED THERMAL POWER, the design limits are met. When using incore detectors to make power distribution maps to determine F_Q and $F^N_{\Delta H}$:~~

~~a. The measurement of total peaking factor, F_Q^{Meas}, shall be increased by (3) percent to account for manufacturing tolerances and further increased by (5) percent to account for measurement error.~~

~~b. The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by (4) percent to account for measurement error.~~

For Condition II events, the core is protected from exceeding ~~(23.5)~~ 19.4 kw/ft locally, and from going below a minimum DNBR of ~~(1.32/~~1.30}, by automatic protection on power, AXIAL POWER IMBALANCE, pressure and temperature. Only conditions 1 through 3, above, are mandatory since the AXIAL POWER IMBALANCE is an explicit input to the Reactor Protection System.

The QUADRANT POWER TILT 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 QUADRANT POWER TILT limit of ~~(4)~~ 3.41% at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. ~~A limiting tilt of (4.5)% can be tolerated before the margin for uncertainty in F_Q is depleted.~~ The limit of ~~(4)~~% was selected to provide an allowance for the uncertainty associated with the indicated power tilt. In the event the tilt is not corrected, the margin for uncertainty on F_Q is reinstated by reducing the power by 2 percent for each percent of tilt in excess of ~~(4)%.~~ 3.41%.

~~3/4.2.5 DNB PARAMETERS~~

~~The limits on the DNB related parameters assure that each of the~~ parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the FSAR initial assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of {1.30} throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 ~~hour basis.~~

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM (RPS) AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

The OPERABILITY of the RPS and ESFAS instrumentation systems ensure that 1) the associated ESFAS action and/or RPS trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for RPS and ESFAS purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

~~The measurement of response time at the specified frequencies~~ provides assurance that the RPS and ESFAS action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such test demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or ~~offsite test measurements or 2) utilizing replacement sensors with certified response times.~~

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. See Bases Figures 3-1 and 3-2 for examples of acceptable minimum incore detector arrangements.

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 so that the response of those features important to safety may be evaluated. 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. ~~This 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. This instrumentation 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 instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of~~

~~REMOTE SHUTDOWN INSTRUMENTATION (Continued)~~

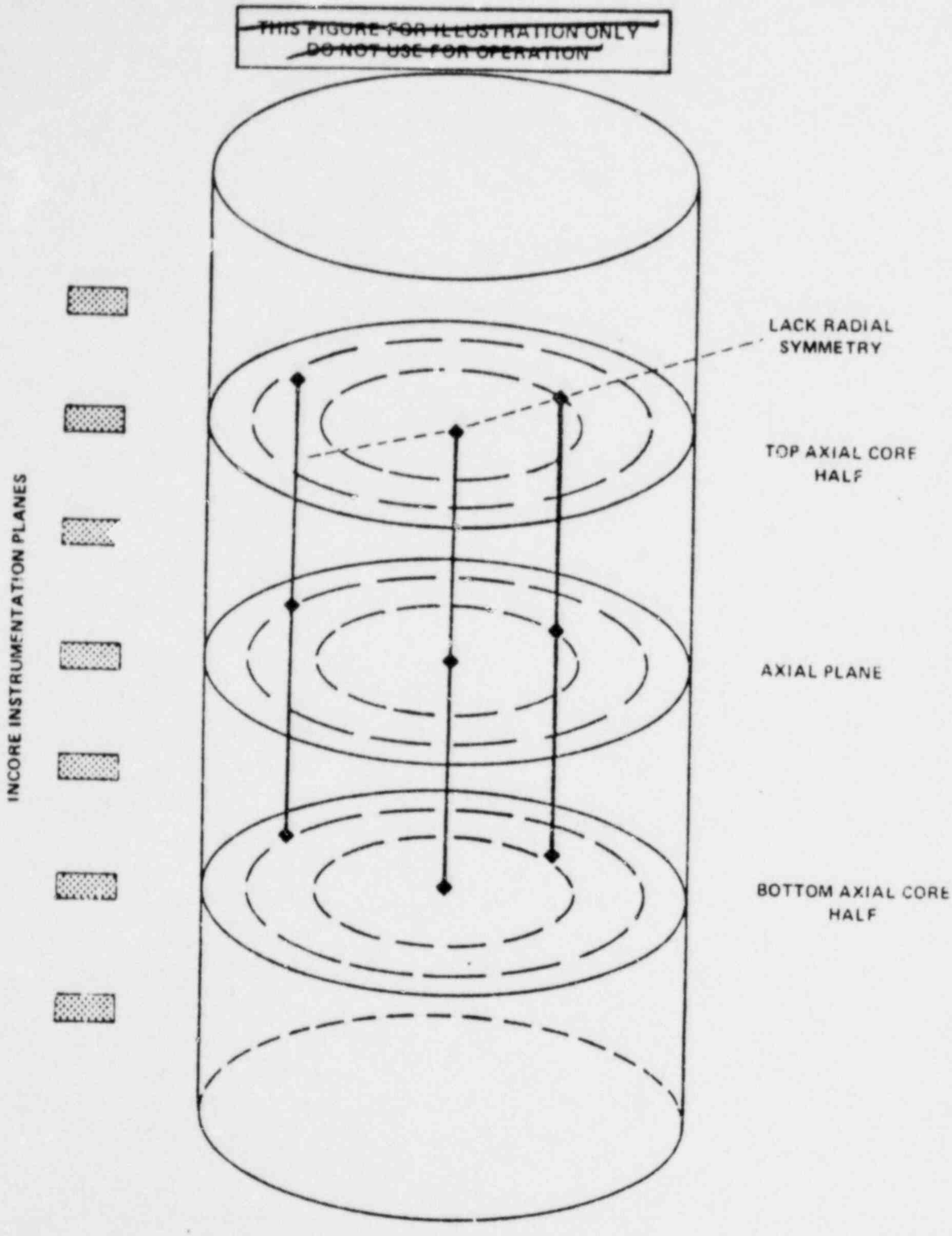
~~HOT STANDBY of the facility from locations outside of the control room~~. ~~This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of Appendix "A"~~, ~~10 CFR 50.~~

~~3/4.3.3.6 POST ACCIDENT INSTRUMENTATION~~

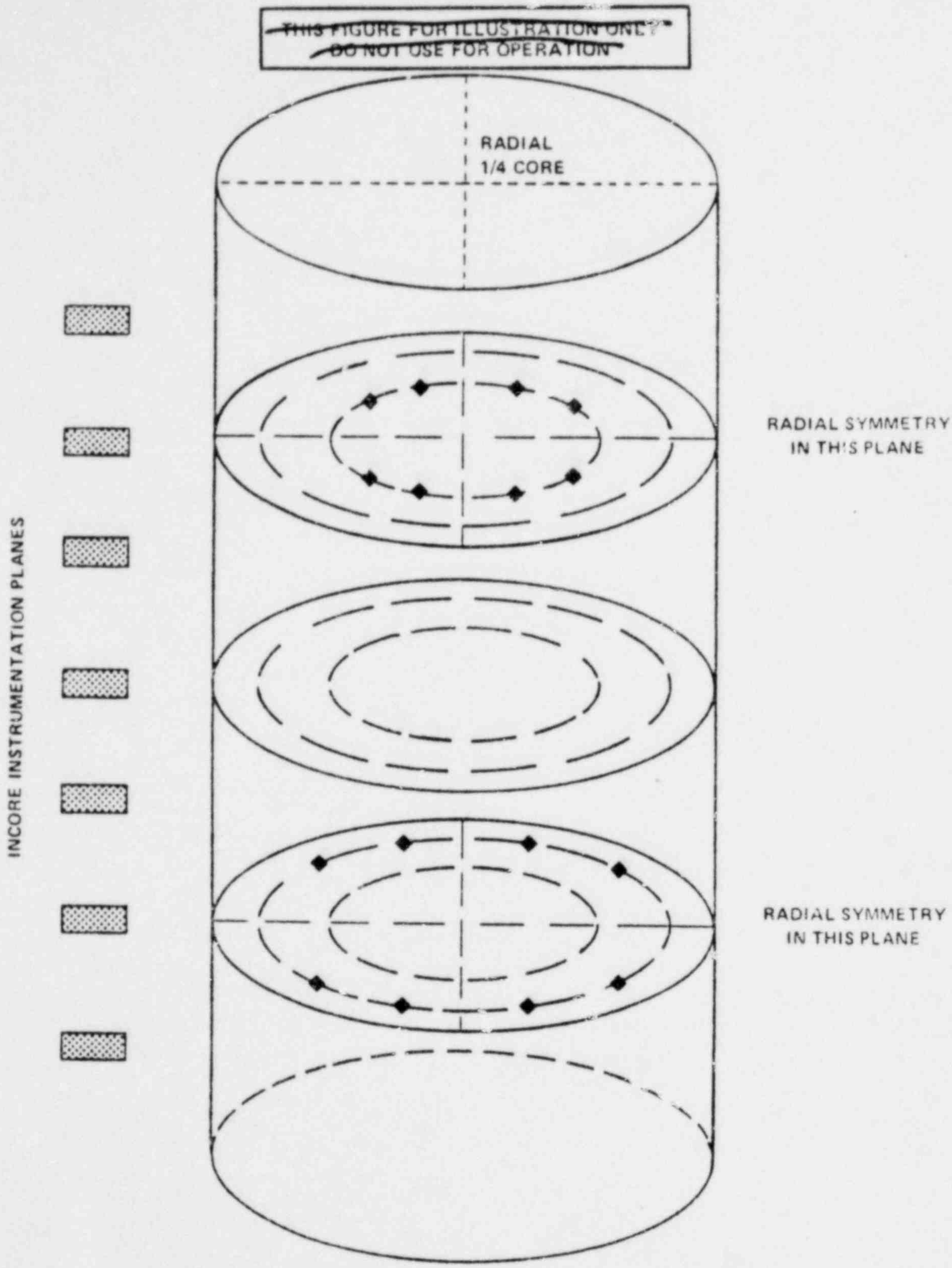
~~The OPERABILITY of the post accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975.~~

~~3/4.3.3.7 CHLORINE DETECTION SYSTEMS~~

~~The OPERABILITY of the chlorine detection systems ensures that an accidental chlorine release will be detected promptly and the control room emergency ventilation system will automatically isolate the control room and initiate its operation in the recirculation mode to provide the required protection. The chlorine detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operations Against an Accidental Chlorine Release," February 1975.~~



Bases Figure 3-1 Incore Instrumentation Specification
Acceptable Minimum AXIAL POWER IMBALANCE Arrangement



Bases Figure 3-2 Incore Instrumentation Specification
Acceptable Minimum QUADRANT POWER TILT Arrangement

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops in operation, and maintain DNBR above ~~(1.32/~~1.30) during all normal operations and anticipated transients. With one reactor coolant pump not in operation in one or both loops, THERMAL POWER is restricted by the Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE and the Nuclear Overpower Based on Pump Monitors trip, ensuring that the DNBR will be maintained above ~~(1.32/~~1.30) at the maximum possible THERMAL POWER for the number of reactor coolant pumps in operation. ~~or the local quality at the point of minimum DNBR equal to (22/15)%, whichever is more restrictive.~~

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a DHR loop into operation in the shutdown cooling mode if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of {2750} psig. Each safety valve is designed to relieve 300,000 lbs per hour of saturated steam at ~~the~~ a pressure ~~valve's setpoint.~~ not greater than 3% above the valve's setpoint.

The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating DHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of {2750} psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from any transient.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief.

The low level limit is based on providing enough water volume to prevent a pressurizer low level or a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the Engineered Safety Feature Actuation System as a result of a reactor scram. The high level limit is based on providing enough steam volume to prevent a pressurizer high level as a result of any transient.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 GPM). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 GPM can be detected by monitoring the secondary coolant. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for {40}% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.~~9.1~~ 12.3 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

~~The steam generator water level limits are consistent with the initial assumptions in the FSAR.~~

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to detect and monitor leakage from the Reactor Coolant Pressure Boundary. ~~These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.~~

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that, while a limited amount of leakage is expected from the RCS, the UNIDENTIFIED LEAKAGE portion of this can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. ~~The (0.5) GPM leakage limit per steam generator ensures that steam generator tube entegrity is maintained in the event of a main steam line rupture or under LOCA conditions~~.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

~~The CONTROLLED LEAKAGE limit of ( ) GPM restricts operation with a total RCS leakage to all RC pump seals in excess of ( ) GPM.~~

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits shown on Table 3.4-1 provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

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

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of the Part 100 limit following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. ~~The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in the specific site parameters of the site, such as site boundary location and meteorological conditions, were not considered in this evaluation. The NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.~~

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. ~~Operation with specific activity levels exceeding 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 10 percent of the units yearly operating time since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.~~

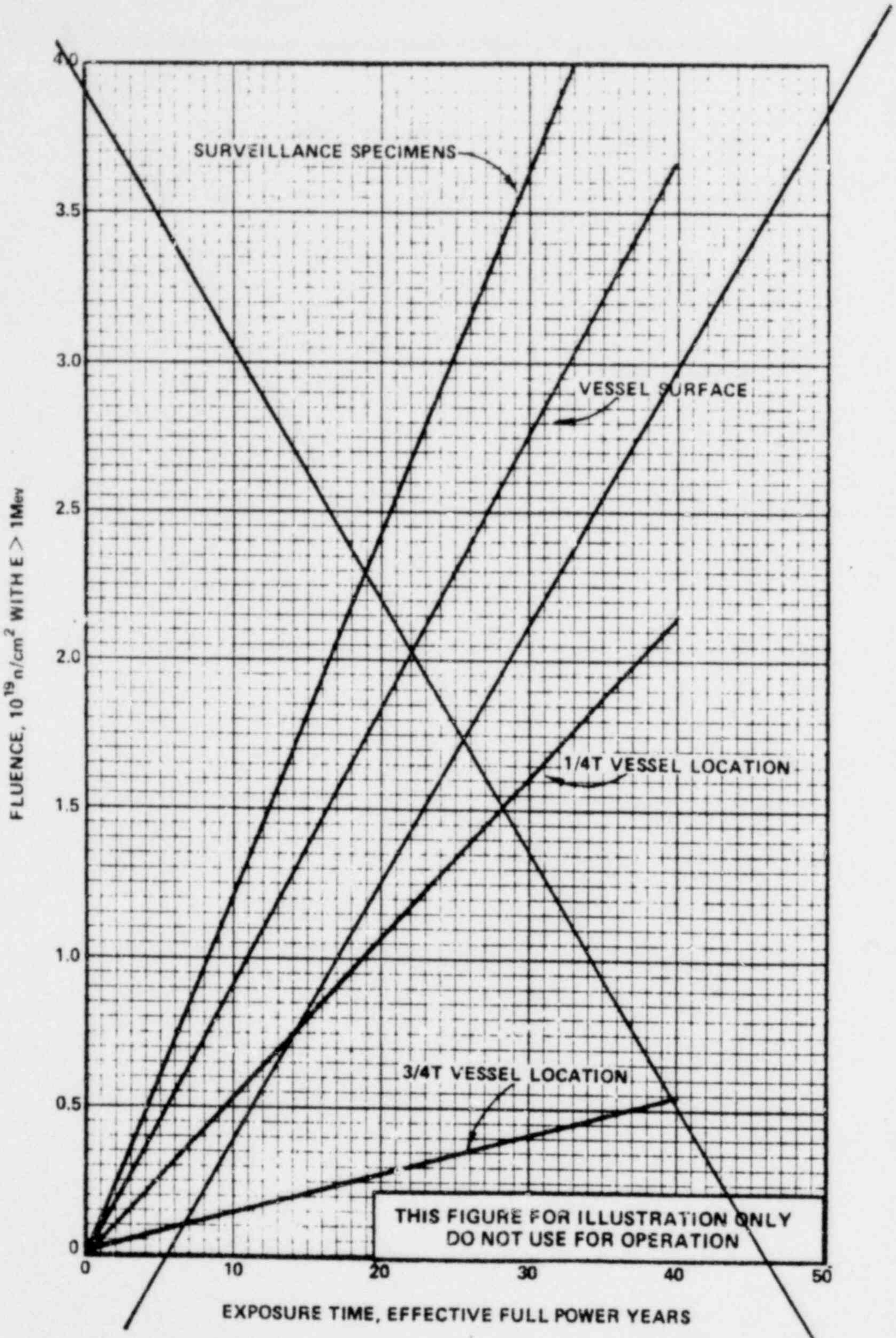
Reducing T_{avg} to < (500)°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. ~~Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.~~

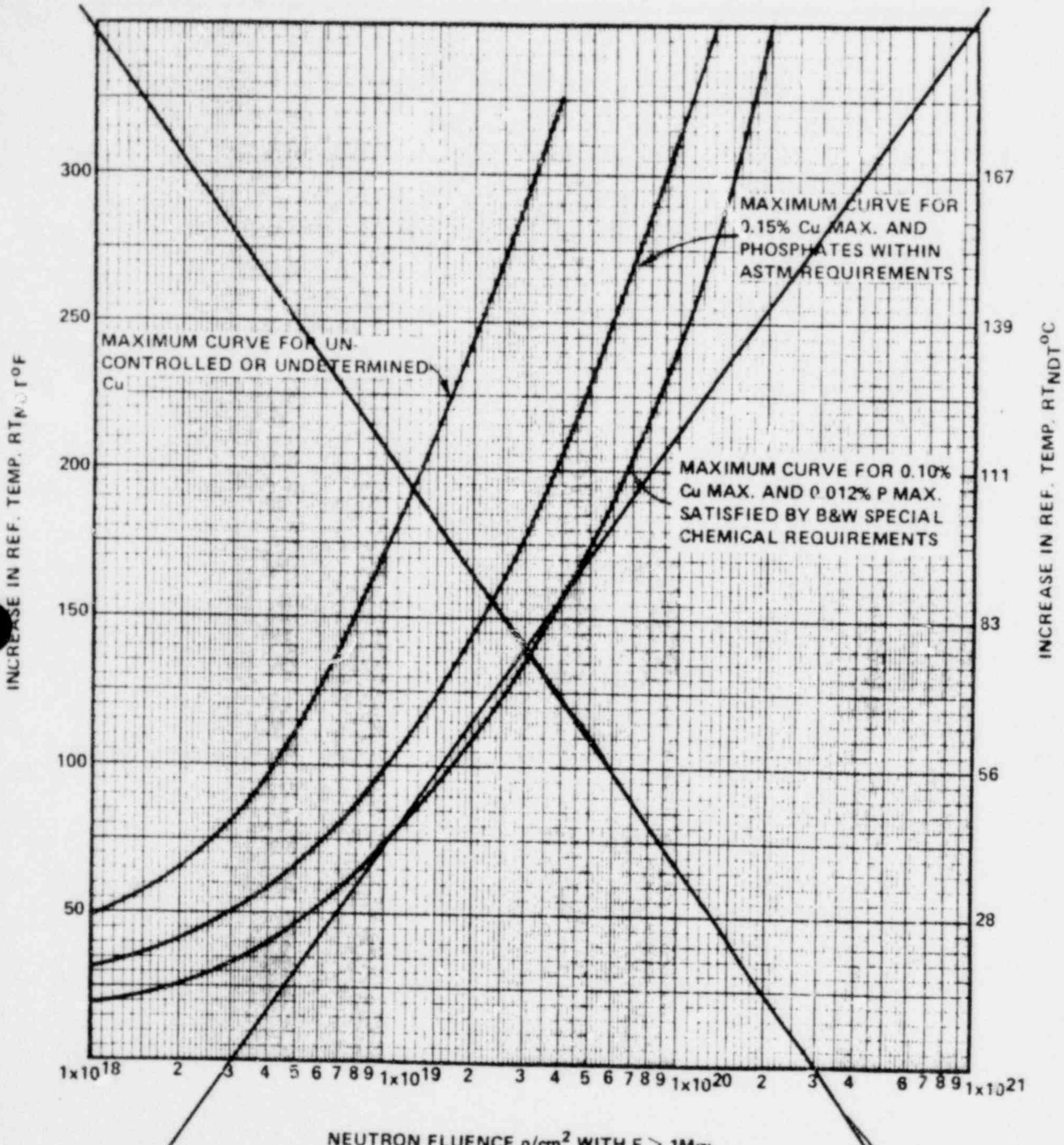
3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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

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



Bases Figure 4-1 Fast Neutron Fluence (E > 1 mev) as a Function of Full Power Service Life



Bases Figure 4-2 Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to 550°F Temperature

~~BASES TABLE 4-1~~

~~REACTOR VESSEL TOUGHNESS~~

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	NDTT F	50 FT-LB/35 MIL TEMP F LONG	50 FT-LB/35 MIL TEMP F TRANS	RT_{NDT} F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS

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

The heatup limit curves, Figures {3.4-2} and {3.4-3}, are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 100°F per hour. The cooldown limit curves, Figures {3.4-2} and {3.4-3} are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of {2} EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT} ~~; the results of these tests are shown in BASES Table 4-1.~~ , but reactor ~~Reactor~~ operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using ~~BASES Figures 4-1 and 4-2.~~ the Babcock Wilcox Topical Report, BAW-10046A Rev. 1. The heatup and cooldown limit curves, of Figures {3.4-2} and {3.4-3} include predicted adjustments for this shift in RT_{NDT} at the end of {2} EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, ~~in accordance with ASTM E185-73,~~ reactor vessel material irradiation surveillance specimens installed ~~near the inside wall of the reactor vessel~~ in the Davis-Besse Unit No. 1 reactor vessel. ~~core area~~. Since the neutron spectra at the irradiation samples and vessel inside the radius are essentially identical, the measured transition shift for a sample can applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure and temperature limits shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components, except steam generator tubes, ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves 1) ensure OPERABILITY, 2) ensure that the valves are not stuck open during normal operation, and 3) demonstrates that the valves ~~begin to open and~~ are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

3/4.5.1 CORE FLOODING TANKS

The OPERABILITY of each core flooding tank ensures that a sufficient volume of borated water will be immediately forced into the reactor vessel in the event the RCS pressure falls below the pressure of the tanks. This initial surge of water into the vessel provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on volume, boron concentration and pressure ensure that the assumptions used for core flooding tank injection in the safety analysis are met.

The tank power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a core flooding tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems with RCS average temperature > ~~(305)~~ 350°F ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the core flooding tanks is capable of supplying sufficient core cooling to maintain the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below ~~(305)~~ 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures, that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Power is required to be removed from any valve which fails to meet single failure criteria. The decay heat removal system leak rate surveillance requirements assure that the leakage rates assumed for the system during the recirculation phase of the low pressure injection will not be exceeded.

3/4.5.4 BORATED WATER STORAGE TANK

The OPERABILITY of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the BWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. The limits on contained water volume, and boron concentration ensure a pH value of between {8.5} and {11.0} of the solution sprayed within ~~containment~~ the reactor building after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

BASES

FOR

SECTION 3/4.6J

CONTAINMENT SYSTEMS SPECIFICATIONS

FOR

BABCOCK AND WILCOX STS

ATMOSPHERIC TYPE CONTAINMENT

REACTOR BUILDING
3/4.6.1 ~~PRIMARY CONTAINMENT~~

REACTOR BUILDING
3/4.6.1.1 ~~CONTAINMENT~~ INTEGRITY

REACTOR BUILDING
~~Primary CONTAINMENT~~ INTEGRITY ensures that the release of radioactive materials from the reactor building ~~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 limits of 10 CFR 100 during accident conditions.

REACTOR BUILDING
3/4.6.1.2 ~~CONTAINMENT~~ LEAKAGE

The limitations on reactor building ~~containment~~ leakage rates ensure that the total reactor building ~~containment~~ leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 59 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ or $\leq 0.75\ L_t$, as applicable, during performance of the periodic tests to account for possible degradation of the reactor building ~~containment~~ leakage barriers between leakage tests.

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

REACTOR BUILDING
3/4.6.1.3 ~~CONTAINMENT~~ AIR LOCKS

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

~~3/4.6.1.4 CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEMS (OPTIONAL)~~

~~The OPERABILITY of the isolation valve and containment channel weld pressurization systems is required to meet the restrictions on overall containment leak rate assumed in the accident analysis. The Surveillance Requirements for determining OPERABILITY are consistent with Appendix "J" of 10 CFR 50.~~

3/4.6.1.5 INTERNAL PRESSURE

The limitations on ~~containment~~ reactor building internal pressure ensure that 1) the reactor building ~~containment structure~~ is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of {3.0} psig and 2) the reactor building ~~containment~~ peak pressure does not exceed the design pressure of ~~(54)~~ 59 psig during LOCA conditions.

The maximum peak pressure obtained from a LOCA event is ~~(45)~~ 53.1 psig. The limit of {3} psig for initial positive ~~containment~~ pressure will limit the total pressure to ~~(48)~~ 56 psig which is less than the design pressure and is consistent with the safety analyses.

~~3/4.6.1.6 AIR TEMPERATURE~~

~~The limitations on containment~~ reactor building ~~average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA.~~

3/4.6.1.7 ~~CONTAINMENT~~ REACTOR BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the ~~containment~~ reactor building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of ~~(48)~~ 56.1 psig in the event of a LOCA. The measurement of ~~containment~~ reactor building tendon lift off force, the visual and metallurgical examination of tendons, anchorages and liner, and the Type A leakage tests are sufficient to demonstrate this capability.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 ~~CONTAINMENT~~ REACTOR BUILDING SPRAY SYSTEM

The OPERABILITY of the ~~containment~~ reactor building spray system ensures that ~~containment~~ reactor building depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower ~~containment~~ reactor build leakage rate are consistent with the assumptions used in the safety analyses. The leak rate surveillance requirements assure that the leakage rates assumed for the system during the recirculation phase will not be exceeded.

3/4.6.2.2 ~~SPRAY ADDITIVE~~ SODIUM HYDROXIDE ADDITIVE SYSTEM ~~(OPTIONAL)~~

The OPERABILITY of the ~~spray additive~~ sodium hydroxide additive system ensures that sufficient NaOH ~~and $Na_2S_2O_3$ are~~ is added to the ~~containment~~ reactor building spray in the event of a LOCA. ~~The minimum $Na_2S_2O_3$ volume and concentration ensures sufficient $Na_2S_2O_3$ is available to remove organic iodine from the containment atmosphere and return it to the spray water.~~ The limits on contained sodium hydroxide solution volume and concentration~~, and contained sodium thiosulfate solution volume and concentration~~ ensure a ~~pH value of between (8.5) and (11.0)~~ stabalized pH of the solution sprayed within ~~containment~~ the reactor building after a design basis accident. The stabalized pH ~~band~~ minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 ~~CONTAINMENT~~ REACTOR BUILDING COOLING SYSTEM ~~(OPTIONAL)~~

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

~~3/4.6.3 IODINE CLEANUP SYSTEM (OPTIONAL)~~

~~The OPERABILITY of the containment iodine cleanup system ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses.~~

3/4.6.4 ~~CONTAINMENT~~ REACTOR BUILDING ISOLATION VALVES

The OPERABILITY of the ~~containment~~ reactor building isolation valves ensures that the ~~containment~~ reactor building atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the ~~containment~~ reactor building atmosphere or pressurization of the ~~containment~~ reactor building. ~~Containment~~ Reactor building isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.5 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within ~~containment~~ the reactor building below its flammable limit during post-LOCA conditions. ~~Either recombiner unit (or~~ The hydrogen purge ~~cleanup~~ system~~)~~ is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within the reactor building. ~~containment.~~ These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

~~The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.~~

3/4.6.6 PENETRATION ROOM ~~EXHAUST AIR CLEANUP~~ VENTILATION SYSTEM ~~(OPTIONAL)~~

The OPERABILITY of the penetration room ~~exhaust air cleanup~~ ventilation system ensures that radioactive materials leaking from the ~~containment~~ reactor building atmosphere through ~~containment~~ reactor building penetrations following a LOCA are filtered and adsorbed prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the LOCA analyses.

~~3/4.6.7 VACUUM RELIEF VALVES (OPTIONAL)~~

~~The OPERABILITY of the primary containment to atmosphere vacuum relief valves ensures that the containment internal pressure does not become more negative than ____ psig. This condition is necessary to prevent exceeding the containment design limit for internal vacuum of ____ psig.~~

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of {1050} psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, (1971) Edition. The total relieving capacity for all valves on all of the steam lines is 12.6 (10^6) lbs/hr which is {1.18} percent of the total secondary steam flow of 10.7 (10^6) lbs/hr at 100% RATED THERMAL POWER. ~~A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.~~

~~STARTUP and/or POWER OPERATION is allowable with safety valves~~ inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Nuclear Overpower channels. The reactor trip setpoint reductions are derived on the following bases:

$$SP = \frac{(X) - (Y)(V)}{X} \times \overset{104.9}{\cancel{(105.5)}}$$

where:

SP = reduced Nuclear Overpower Trip Setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam generator
= 2

104.9 ~~(105.5)~~ = Nuclear Overpower Trip Setpoint specified in Table 2.2.1

X = Total relieving capacity of all safety valves per steam generator in lbs/hour
= 12 624 784 lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour
= 1610 592 lbs/hour

3/4.7.1.2 ~~AUXILIARY~~ EMERGENCY FEEDWATER SYSTEMS

The OPERABILITY of the ~~auxiliary~~ emergency feedwater systems ensures that the Reactor Coolant System can be cooled down to less than ~~(305)~~ 280°F from normal operating conditions in the event of a total loss of offsite power.

~~Each electric~~ The motor driven ~~auxiliary~~ EMERGENCY feedwater pump is capable of delivering a total feedwater flow of ~~(350)~~ 780 gpm at a pressure of ~~(1133)~~ 1126 psig to the entrance of the steam generators. Each steam driven ~~auxiliary~~ EMERGENCY feedwater pump is capable of delivering a total feedwater flow of ~~(700)~~ 780 gpm at a pressure of ~~(1133)~~ 1126 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than ~~(305)~~ 280°F where the Decay Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than ~~(305)~~ 280°F in the event of a total loss of offsite power or of the main feedwater system. The minimum water volume is sufficient to maintain the RCS at HOT ~~STANDBY~~ SHUTDOWN conditions for {4.5} hours with steam discharge to atmosphere concurrent with loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM ~~LINE ISOLATION~~ BLOCK VALVES

The OPERABILITY of the main steam ~~line isolation~~ block valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within ~~containment~~ in the event the steam line rupture occurs within ~~containment.~~ The OPERABILITY of the main steam ~~isolation~~ valves within the closure times of the surveillance requirements are consistent with the assumptions used in the safety analyses.

~~3/4.7.1.6 SECONDARY WATER CHEMISTRY~~

~~Maintaining the steam generator feedwater within the~~ limits of ~~this Specification will~~ control the introduction ~~of potentially~~ corrosive impurities into the steam ~~generators and~~ ~~to~~ minimize tube degradation.

~~3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION~~

~~The limitation on steam generator pressure and temperature ensures~~ that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of (70)°F and (200) psig are based on a steam generator RT_{NDT} of ( )°F and are sufficient to prevent brittle fracture.

~~3/4.7.3 COMPONENT COOLING WATER SYSTEM~~

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is ~~consistent with the assumptions used in the safety analyses.~~

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 EMERGENCY COOLING POND ~~ULTIMATE HEAT SINK (OPTIONAL)~~

The limitations on the emergency cooling pond depth ~~ultimate heat sink level~~ and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum indicated depth ~~water level~~ and maximum temperature are based on providing a 30 day cooling water supply to safety related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants", March 1974.

~~3/4.7.6 FLOOD PROTECTION (OPTIONAL)~~

~~The limitation on flood protection ensures that facility protective actions will be taken (and operation will be terminated) in the event of flood conditions. The limit of elevation (__) Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment.~~

3/4.7.7 CONTROL ROOM EMERGENCY AIR CONDITIONING AND AIR FILTRATION ~~CLEANUP~~ SYSTEM

The OPERABILITY of the control room emergency air conditioning and air filtration ~~cleanup~~ system ensures that ~~1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2)~~ the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix "A", 10 CFR 50.

~~3/4.7.8 ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM (OPTIONAL)~~

~~The OPERABILITY of the ECCS pump room exhaust air cleanup system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations were assumed in the safety analyses.~~

3/4.7.9 HYDRAULIC ~~SNUBBERS~~ SHOCK SUPPRESSORS (SNUBBERS)

The hydraulic ~~snubbers~~ shock suppressors are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. The only ~~snubbers~~ shock suppressors excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety related system.

The inspection frequency ~~applicable to snubbers containing seals fabricated from materials which have been demonstrated compatible with their operating environment~~ is based upon maintaining a constant level of ~~snubber~~ shock suppressor protection. Therefore, the required inspection interval varies inversely with the observed ~~snubber~~ shock suppressor failures. The number of inoperable ~~snubbers~~ shock suppressors found during an inspection of these ~~snubbers~~ shock suppressors determines the time interval for the next required inspection. ~~of these snubbers.~~ Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide further assurance of ~~snubber~~ shock suppressor reliability, a representative sample of the installed ~~snubbers~~ shock suppressors will be functionally tested during plant shutdowns at 18 month intervals. These tests will include stroking of the ~~snubbers~~ shock suppressors to verify proper piston movement, lock-up and bleed. Observed failures of these sample ~~snubbers~~ shock suppressors will require functional testing of additional units. To minimize personnel exposures, ~~snubbers~~ shock suppressors installed in high radiation zones or in especially difficult to remove locations may be exempted from these functional testing requirements provided the OPERABILITY ~~of these snubbers~~ was demonstrated during functional testing at either the completion of their fabrication or at a subsequent date.

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

BASES

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one of each of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

~~The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the facility status~~.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volumes having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analysis.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 REACTOR BUILDING ~~CONTAINMENT~~ PENETRATIONS

The requirements on reactor building ~~containment~~ penetration closure and OPERABILITY ensure that a release of radioactive material within the reactor building ~~containment~~ will be restricted from leakage to the environment. The OPERABILITY and closure requirements are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of ~~containment~~ reactor building pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

~~3/4.9.6 FUEL HANDLING BRIDGE OPERABILITY~~

~~The OPERABILITY requirements of the hoist bridges used for movement of fuel assemblies ensures that: 1) fuel handling bridges will be used for movement of control rods and fuel assemblies, 2) each hoist has sufficient load capacity to lift a fuel element, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.~~

3/4.9.7 CRANE TRAVEL - AUXILIARY ~~SPENT FUEL STORAGE~~ BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 COOLANT CIRCULATION

The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

~~REACTOR BUILDING~~ ~~FILTRATION~~
~~3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM~~

~~The OPERABILITY of this system ensures that the~~ ~~reactor building~~ ~~containment purge and exhaust penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.~~ ~~reactor building~~

~~3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL WATER LEVEL~~

~~The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.~~

BASES

FUEL HANDLING AREA
~~SPENT FUEL~~ VENTILATION
3/4.9.12 ~~STORAGE POOL AIR CLEANUP~~ SYSTEM

~~spent~~ fuel handling area ventilation
The requirements on the ~~storage pool air cleanup~~ system to be operating or OPERABLE ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their specified group heights and insertion limits and to be assigned to other than specified control rod groups, and permits AXIAL POWER IMBALANCE and QUADRANT POWER TILT limits to be exceeded during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth, 2) determine the reactor stability index and damping factor under xenon oscillation conditions and 3) calibrate AXIAL POWER IMBALANCE and QUADRANT POWER TILT instrumentation.

3/4.10.2 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER and is required to verify the fundamental nuclear characteristics of the reactor core and related instrumentation.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under various flow conditions and is required in order to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.5 MINIMUM TEMPERATURE FOR CRITICALITY

This special test exception permits reactor criticality at low THERMAL POWER levels with T_{avg} below 525°F during PHYSICS TESTS which provide data that can be used to verify the adequacy of design codes for new fuel designs (16 x 16 first of a kind fuel assembly) for reduced temperature conditions.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area is shown on Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone is shown on Figure 5.1-2.

5.2 ~~CONTAINMENT~~ REACTOR BUILDING

CONFIGURATION

5.2.1 The reactor ~~containment~~ building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 116 feet.

b. Nominal inside height = 207 feet.

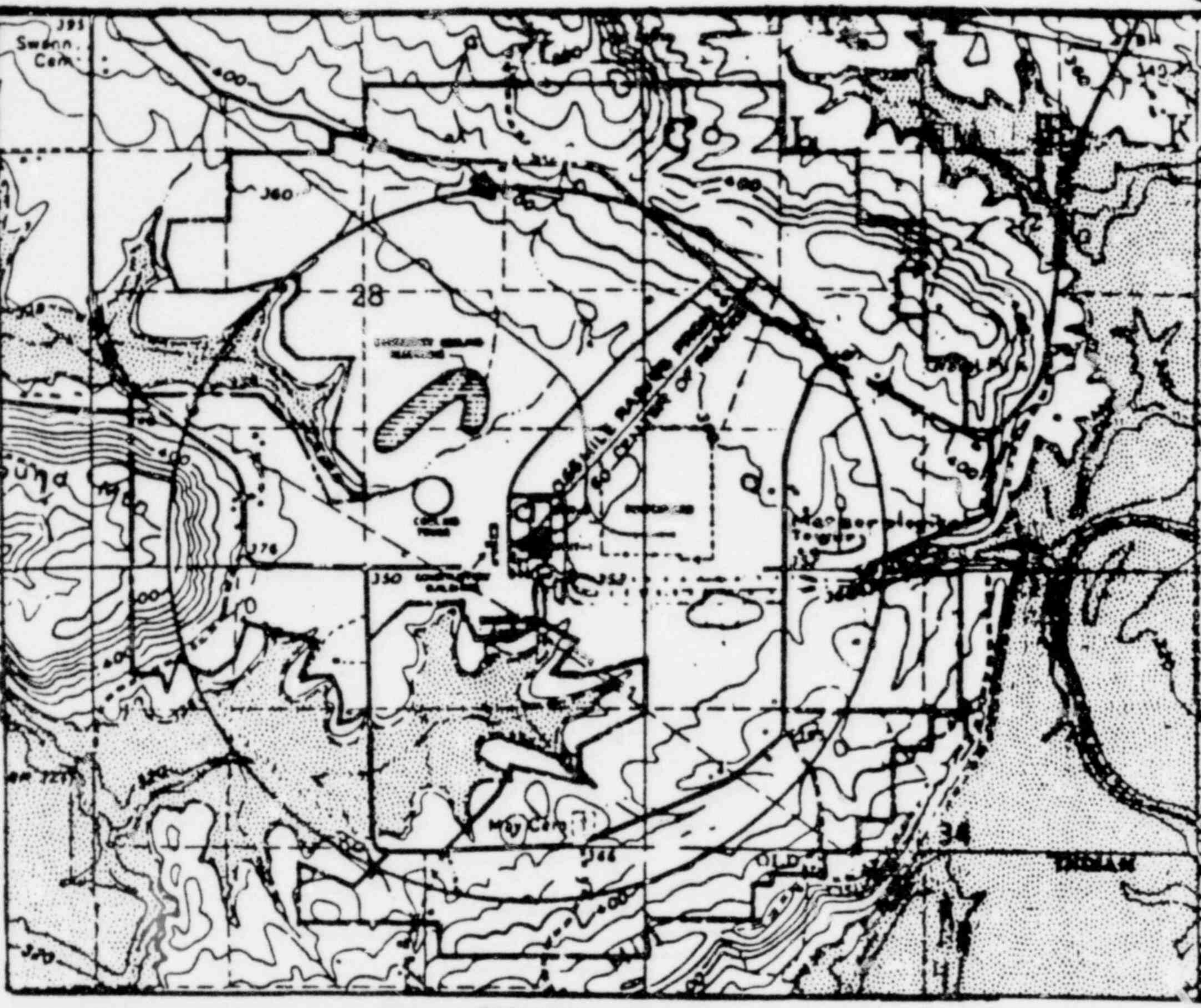
c. Minimum thickness of concrete walls = 3.75 feet.

d. Minimum thickness of concrete roof = 3.25 feet.

e. Minimum thickness of concrete floor pad = 9.0 feet.

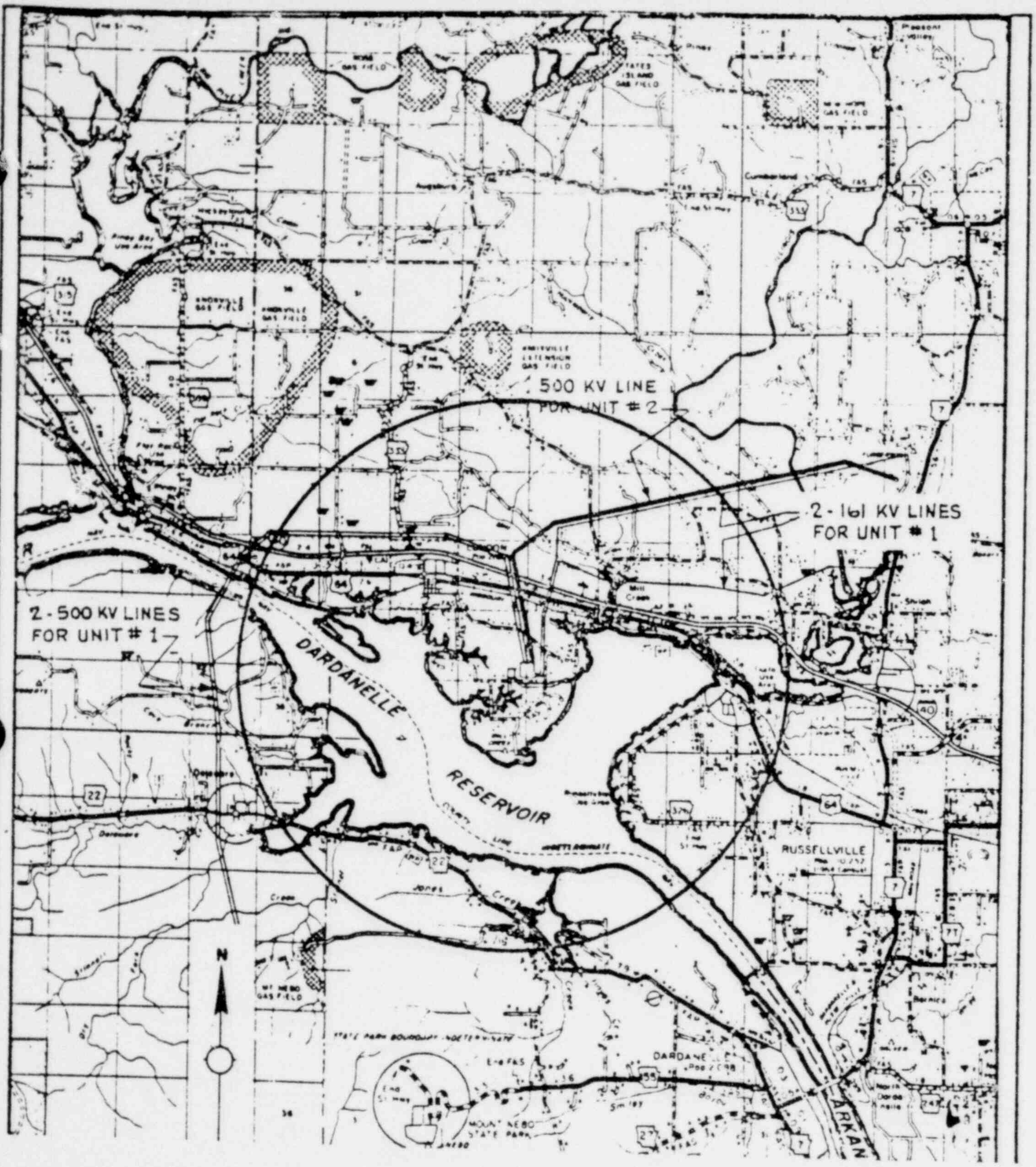
f. Nominal thickness of steel liner = 0.25 inches.

g. Net free volume = 1.85(10⁶) cubic feet.



EXCLUSION AREA

FIGURE 5.1-1



LOW POPULATION ZONE

FIGURE 5.1-2

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 59 psig and a temperature of 286°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2271 grams uranium. ~~The initial core loading shall have a maximum enrichment of ___ weight percent U-235.~~ Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

~~The first cycle fuel loading shall contain ___ burnable poison rod assemblies with each assembly containing up to ___ burnable poison rods of sintered Al_2O_3-B_4C clad with Zircaloy-4.~~

CONTROL RODS

5.3.2 The reactor core shall contain 61 safety and regulating and 8 axial power shaping (APSR) control rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber material. The APSR's shall contain a nominal 36 inches of absorber material at their lower ends. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.1 ~~(5.2)~~ of the FSAR, with allowance for normal degradation pursuant to applicable Surveillance Requirements.

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer and pressurizer surge line which is 670 °F.

VOLUME

5.4.2 The total ~~water and steam~~ volume of the reactor coolant system is 11,800 cubic feet. ~~at a nominal T_{avg} of (525)°F.~~

~~5.5 METEOROLOGICAL TOWER LOCATION~~

~~5.5.1 The meteorological tower shall be located as shown on Figure (5.1.1).~~

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new ~~and spent~~ fuel ~~storage~~ racks are designed and shall be maintained with a nominal 21.0 inch center-to-center distance between fuel assemblies placed in the ~~storage~~ racks to ensure a k_{eff} equivalent to $\leq$ 0.90 with the ~~storage~~ pool filled with unborated water. ~~The k_{eff} of $\leq$ 0.95 includes a conservative allowance of (3.3)% Δk/k for uncertainties as described in Section (9.1) of the FSAR.~~ The spent fuel racks are designed and shall be maintained with a nominal 13.5 inch center to center distance between fuel assemblies placed in the racks to ensure a k_{eff} equivilant to ≤ 0.95 with the pool filled with unborated water.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 392 ft.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 590 fuel assemblies.

~~5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT~~

~~5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limit of Table 5.7-1~~.

~~TABLE 5.7-1~~

~~COMPONENT CYCLIC OR TRANSIENT LIMITS~~

Component or System	Cycle or Transient Limit	Design Cycle or Transient
1. Reactor Coolant System	(240) Heatup and Cooldown Cycles	(70°F to 557°F to 70°F)
2. Reactor Coolant System	(160) Step Load Reduction Cycles (Resulting from turbine trip)	(100% to 8% RTP*)
3. Reactor Coolant System	(150) Step Load Reduction Cycles (Resulting from electrical load rejection)	(100% to 8% RTP*)
4. Reactor Coolant System	(40) Reactor Trip Cycles (Resulting from loss of electric power to all RC pumps)	Reactor Trip
5. Reactor Coolant System	(160) Reactor Trip Cycles (Resulting from turbine trip without automatic control action)	Reactor Trip
6. Reactor Coolant System	(40) Reactor Trip Cycles (Resulting from rod withdrawal accident)	Reactor Trip
7. Once Through Steam Generator	(88) Reactor Trip Cycles (Resulting from complete loss of all main feedwater)	Reactor Trip
8. Once Through Steam Generator	(40) Reactor Trip Cycles (Resulting from loss of station power)	Reactor Trip
9. Once Through Steam Generator	(20) Reactor Trip Cycles (Resulting from loss of feedwater to one steam generator)	Reactor Trip

*RATED THERMAL POWER

~~TABLE 5.7-1 (Continued)~~

~~Component or System~~	~~Cycle or Transient Limit~~	~~Design Cycle or Transient~~
10. Once Through Steam Generator	(10) Reactor Trip Cycles (Resulting from stuck open turbine bypass valve)	Reactor Trip
11. Reactor Coolant System	(80) Rapid Depressurization	(2200 psig to 300 psig in one hour)
12. Reactor Coolant System	(20) Change of Flow Cycles	Loss of one or more RC pumps
13. Reactor Coolant System	(20) Hydrostatic Test	Pressurized to $\geq$ (3125) psig
14. Once Through Steam Generator	(35) Hydrostatic Tests	Pressurized to $\geq$ (3125) psig
15. Reactor Coolant System	(480) Test Transients	High Pressure Injection Test
16. Reactor Coolant System	(240) Test Transients	Core Flooding Check Valve Test