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FACILITY OPERATING LICENSES DPR-70 AND DPR-75
DOCKET NOS. 50-272 AND 50-311
CHANGE TO TECHNICAL SPECIFICATIONS
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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 9.0% of narrow range instrument span-each steam generator	≥ 8.0% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 10.0% of narrow range instrument span--each steam generator	≤ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 9.0% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pumps	≥ 2900 volts-each bus	≥ 2850 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 56.5 Hz - each bus	≥ 56.4 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	≥ 45 psig	≥ 45 psig
B. Turbine Stop Valve Closure	≤ 15% off full open	≤ 15% off full open
18. Safety Injection Input from ~~SSPS~~ ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) (T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

T' = Indicated T_{avg} at RATED THERMAL POWER $\leq$ 577.9°F

P = Pressurizer pressure, psig

P' = 2235 psig (indicated RCS nominal operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead lag controller for T_{avg} τ_1 = 30 secs, τ_2 = 4 secs.

S = Laplace transform operator, Sec^{-1}

BASES

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + 0.3(1-P)]$$

where P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods FULLY WITHDRAWN to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping and fittings are designed to ANSI B 31.1 1955 Edition while the valves are designed to ANSI B 16.5, MSS-SP-66-1964, or ASME Section III-1968, which permit maximum transient pressures of up to 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ 1.6% Δk/k###.

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

ACTION:

With the SHUTDOWN MARGIN < 1.6% Δk/k###, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥ 1.6% Δk/k###:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that control ~~bank withdrawal is~~ banks are within the limits of Specification 3.1.3.5.

c. When in MODE 2##, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5.

*See Special Test Exception 3.10.1

#With K_{eff} ≥ 1.0

##With K_{eff} < 1.0

1.88% delta k/k during Cycle 11 of operation.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0 delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition.

b. Less negative than -4.4×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.4.a - MODES 1 and 2* only#
Specification 3.1.1.4.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.4.a, above, operations in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 delta k/k/°F within 24 hours or be ~~in HOT STANDBY~~ within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification ~~3.1.3.6.~~ 3.1.3.5.

2. The control rods are maintained within the withdrawal limits established abpve until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.4.b, above, be in HOT SHUTDOWN within 12 hours.

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

#See Special Test Exception 3.10.3

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by specifications 3.1.2.1 and 3.1.2.2:

a. A boric acid storage system with:

1. A contained volume of borated water in accordance with figure 3.1.2,

2. A boron concentration in accordance with figure 3.1-2 ~~3.1.2,~~ and

3. A minimum solution temperature of 63°F.

b. The refueling water storage tank with:

1. A contained volume of between 364,500 and 400,000 gallons of water,

2. A boron concentration of between 2,300 and 2,500 ppm, and

3. A minimum solution temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required boration water systems, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta K/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status 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

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position within one hour after rod motion.

APPLICABILITY: MODES 1* and 2*

ACTION:

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

b. With more than one full length rod inoperable or mis-aligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or mis-aligned from its group step counter demand position by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

1. The rod is restored to OPERABLE status within the above alignment requirements, or

2. The remainder of the rods in the bank with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of
3.1-3
Figures 3.1-1 and ~~3.1-2,~~ the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation, or

3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

~~POSITION INDICATION SYSTEM SHUTDOWN~~ SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be FULLY WITHDRAWN.

APPLICABILITY: MODES 1*, and 2*#@

ACTION:

With a maximum of one shutdown rod not FULLY WITHDRAWN, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. FULLY WITHDRAW the rod, or,

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be FULLY WITHDRAWN by use of the group demand counters, and verified by the analog rod position indicators**:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3

**For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

#With Keff greater than or equal to 1.0

@Surveillance 4.1.3.4.a is applicable prior to withdrawing control banks in preparation for startup (Mode 2).

~~POSITION INDICATION SYSTEM SHUTDOWN~~ & CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and ~~3.1-2~~ 3.1-3.

APPLICABILITY: MODES 1*, and 2*#

Change "nn"

Change "b"

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figures, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours by use of the group demand counters and verified by the analog rod position indicators** except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours**.

*See Special Test Exceptions 3.10.2 and 3.10.3

**For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

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

FIGURE 3.1-~~2~~3

INTENTIONALLY LEFT BLANK PENDING
COMMISSION APPROVAL OF THREE LOOP OPERATION

FIGURE 3.1-~~2~~3 ROD BANK INSERTION LIMITS VERSUS THERMAL POWER FOR THREE LOOP OPERATION

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the above limits and ACTION a.2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the above limits for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel·

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the limits of Specification 3.2.1. Penalty deviation outside of the limits shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the limits at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the limits at THERMAL POWER levels below 50% of RATED THERMAL POWER.

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SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to 4.2.1.3 above or by linear interpolation between the most recently ~~measured~~ value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

measured

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	R(9)	1, 2, and *
2. Power Range, Neutron Flux	S	D(2), M(3) and Q(6)	Q	1, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	Q	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(6)	Q	1, 2
5. Intermediate Range, Neutron Flux	S	R(6)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	S(7)	R(6)	Q and S/U(1)	2, 3, 4, 5 and *
7. Overtemperature ΔT	S	R	Q	1, 2
8. Overpower ΔT	S	R	Q	1, 2
9. Pressurizer Pressure--Low	S	R	Q	1, 2
10. Pressurizer Pressure--High	S	R	Q	1, 2
11. Pressurizer Water Level--High	S	R	Q	1, 2
12. Loss of Flow - Single Loop	S	R	Q	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow Two Loops	S	R	N.A.	1
14. Steam Generator Water Level--Low-Low	S	Change "gg" R	Q	1, 2
15. Deleted				
16. Undervoltage - Reactor Coolant Pumps	N.A.	R	Q	1
17. Underfrequency - Reactor Coolant Pumps	N.A.	R	Q	1
18. Turbine Trip				
a. Low Autostop Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
b. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)(5)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	~~N.A.~~ 1
21. Reactor Trip Breaker	N.A.	N.A.	M(5)(11)(13) and R(14)	1, 2 and *
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2 and *

Change "i"

TABLE 3.3-3 (Continued)

ACTION 19 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours.

b. The Minimum Channels OPERABLE requirements is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.1.

ACTION 20 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 6 hours or, be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.1 provided the other channel is OPERABLE.

ACTION 21 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may proceed provided that the inoperable channel is restored to OPERABLE within 72 hours.

ACTION 22 - NOT USED

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within ~~the~~ 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

TABLE 4.3-2 (Continued)

TABLE NOTATION

* Outputs are up to, but not including, the output relays.

** The provisions of Specification 4.0.4 are not applicable.

(1) Each logic channel shall be tested at least once per 62 days on a STAGGERED TEST BASIS. The CHANNEL FUNCTION TEST of each logic channel shall verify that its associated diesel generator automatic load sequence timer is OPERABLE with the interval between each load block within 1 second of its design interval.

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

(3) The CHANNEL FUNCTIONAL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

(4) NOT USED

(5) NOT USED

(6) Inputs from Undervoltage, Vital Bus, shall be tested monthly. Inputs from Solid State Protection System shall be tested every 62 days on a STAGGERED TEST BASIS.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

SOURCE CHECK,

TABLE 3.3-6 (Continued)
RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	$\leq 3.0 \times 10^{-2} \mu Ci/cm^3$ (Alarm only)	10^{-3}-$10^{1} \mu Ci/cm^3$	23
2) High Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	$\leq 1.0 \times 10^{2} \mu Ci/cm^3$ (Alarm only)	10^{-1}-10^{5} $\mu Ci/cm^3$	23
3) Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps)	1/ MS Line	1,2,3&4	$\leq$10 mR/hr (Alarm only)	1-10^{4} mR/hr	23
4) Condenser Exhaust System	1	1,2,3&4	$\leq 1.27 \times 10^{4}$ cpm (Alarm only)	1-10^{6} cpm	23
3. CONTROL ROOM					
a. Air Intake - Radiation Level	2/Intake##	**	$\leq 2.48 \times 10^{3}$ cpm	10^{1}-10^{7} cpm	24, 25

Change "cc"

Control Room air intakes shared between Unit 1 and 2.

** ALL MODES and during movement of irradiated fuel assemblies and during ~~core alterations~~ CORE ALTERATIONS.

Change "uu"

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 23 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

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

ACTION 24 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel(s) to OPERABLE status within 7 days or initiate and maintain operation of the Control Room Emergency Air Conditioning System (CREACS) in the pressurization or recirculation mode of operation. CORE ALTERATIONS and movement of irradiated fuel assemblies will be suspended during operation in the recirculation mode.

ACTION 25 – With no channels OPERABLE in a Control Room air intake, immediately initiate and maintain operation of the CREACS in the pressurization or recirculation mode of operation. CORE ALTERATIONS and movement of irradiated fuel assemblies will be suspended during operation in the recirculation mode.

TABLE 4.[illegible] (Continued)
RADIATION MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNELS CHECKS	SOURCE CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	S	M	R	Q	1, 2, 3 & 4
2) High Range Auxiliary Building Exhaust System (Plant Vent)	S	M	R	Q	1, 2, 3 & 4
3) Main Steamline Discharge (Safety Valves and Atmospheric Dumps)	S	M	R	Q	1, 2, 3 & 4
4) Condenser Exh. Sys.	S	M	R	Q	1, 2, 3 & 4
3. CONTROL ROOM					
a. Air Intake - Radiation Level	S	M	R	Q	**

** ALL MODES and during movement of irradiated fuel assemblies and during ~~core alterations.~~ CORE ALTERATIONS

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be operable.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. As shown in Table 3.3-11.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, ~~AND~~ a CHANNEL CALIBRATION, operations at the frequencies shown in Table 4.3-11.

and CHANNEL FUNCTIONAL TEST

TABLE 3.3-11 (continued)

TABLE NOTATION

ACTION 1 With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 2 With the number of OPERABLE accident monitoring channels less than the MINIMUM Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

change "ww"

ACTION 3 deleted

ACTION 4 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table ~~3.3-11,~~ 3.3-11 operation may proceed provided that an OPERABLE Steam Generator Wide Range Level channel is available as an alternate means of indication for the Steam Generator with no OPERATABLE Auxiliary Feedwater Flow Rate channel.

ACTION 5 ~~With the number of OPERABLE channels less than the Required Number of Channels shown in Table 3.3-11, operation may proceed provided that Steam Tables are available in the Control Room and the following Required Channels shown in Table 3.3-11 are OPERABLE to provide an alternate means of calculating Reactor Coolant System subcooling margin:~~

~~a. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)~~

~~b. Reactor Coolant Pressure (Wide Range)~~

deleted

change "t"

TABLE 4.3-11

SURVEILLANCE REQUIREMENTS FOR

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R	~~NA~~ N.A.
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R	~~NA~~ N.A.
3. Reactor Coolant Pressure (Wide Range)	M	R	~~NA~~ N.A.
4. Pressurizer Water Level	M	R	~~NA~~ N.A.
5. Steam Line Pressure	M	R	
6. Steam Generator Water Level (Narrow Range)	M	R	~~NA~~ N.A.
7. Steam Generator Water Level (Wide Range)	M	R	~~NA~~ N.A.
8. Refueling Water Storage Tank Water Level	M	R	~~NA~~ N.A.
9. deleted			
10. Auxiliary Feedwater Flow Rate	~~SU#~~ S/U #	R	~~NA~~ N.A.
11. Reactor Coolant System Subcooling Margin Monitor	M	~~N/A~~ N.A.*	~~NA~~ N.A.

Change "hh"

Change "u"

#Auxiliary Feedwater System is used on each startup and flow rate indication is verified at that time.

*The instruments used to develop RCS subcooling margin are calibrated on an 18 month cycle; the monitor will be compared quarterly with calculated subcooling margin for known input values.

TABLE 4.3-11 (Continued)
SURVEILLANCE REQUIREMENTS FOR ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
12. PORV Position Indicator	M	~~NA~~ N.A.	R
13. PORV Block Valve Position Indicator	M	~~NA~~ N.A.	Q*
14. Pressurizer Safety Valve Position Indicator	M	~~NA~~ N.A.	R
15. Containment Pressure - Narrow Range	M	~~NA~~ R	~~NA~~ N.A.
16. Containment Pressure - Wide Range	M	R	~~NA~~ N.A.
17. Containment Water Level - Wide Range	M	R	~~NA~~ N.A.
18. Core Exit Thermocouples	M	R	~~NA~~ N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)	M	R	~~NA~~ N.A.

Change "bb"

Change "hh"

* Unless the block valve is closed in order to meet the requirements of Action b, or c in specification 3.4.3.

TABLE 4.3-12 (Continued)

TABLE NOTATION

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

1. Instrument indicates measured levels at or above the alarm/trip setpoint.
2. Circuit failure. (Loss of Power)
3. Instrument indicates a downscale failure. (Indication on instrument drawer in Control Equipment Room only)

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

1. Instrument indicates measured levels at or above the alarm/trip setpoint.
2. Circuit failure. (Loss of Power)
3. Instrument indicates a downscale failure. (Indication on instrument drawer in Control Equipment Room only)
4. Instrument controls not set in operate mode. (On instruments equipped with operate mode switches only)

(3) The initial CHANNEL CALIBRATION was performed using appropriate liquid or gaseous calibration sources obtained from reputable suppliers. The activity of the calibration sources were reconfirmed using a multi-channel analyzer which was calibrated using one or more NBS standards.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

* During liquid additions to the tank.

** If tank level indication is not provided, vertification will be done by visual inspection.

\# The R18 channel is an in-line channel which requires periodic decontamination. Any count rate indication above 10,000 cpm constitutes a SOURCE ~~CHANNEL~~ CHECK for compliance purposes.

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL~~S~~ ~~OPERABLE~~ CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	*
b. Oxygen Monitor	D	N.A.	Q(4)	M	**
2. CONTAINMENT PURGE AND PRESSURE - VACUUM RELIEF					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	***
3. PLANT VENT HEADER SYSTEM#					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	N.A.	*
e. Sampler Flow Rate Monitor	W	N.A.	R	N.A.	*

The following process streams are routed to the plant vent where they are effectively monitored by the instruments described:

(a) Condenser Air Removal System
(b) Auxiliary Building Ventilation System
(c) Fuel Handling Building Ventilation System
(d) Radwaste Area Ventilation System
(e) Containment Purges

4.3.4.3 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

b. At least once ~~one~~ per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

4.3.4.4 Verify the test frequency maintains the probability of a missile ejection incident within NRC guidelines by reviewing the methodology presented in WCAP-11525:

a. At least once every two refueling outages.

b. After modifications to the main turbine or turbine overspeed protection valves.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.4 Two# residual heat removal loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5.##

ACTION:

a. With less than the above required loops operable, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

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

SURVEILLANCE REQUIREMENTS

4.4.1.4 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

\# One RHR loop may be inoperable for up to two hours for surveillance testing, provided the other RHR loop is OPERABLE and in operation. Additionally, four filled reactor coolant loops, with at least two steam generators with their secondary side water levels greater than or equal to 5% (narrow range), may be substituted for one residual heat removal loop.

\## A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 312°F unless 1) the pressurizer water volume is less than 1650 cubic feet (equivalent to 93.2% approximately ~~92%~~ of level), or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

* Systems supporting RHR loop operability may be excepted as follows:

a. The normal or emergency power source may be inoperable.

b. One service water header may be out of service provided the equipment listed in Table 3.4-3 is OPERABLE.

** The residual heat removal pumps may be de-energized for up to 2 hours provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

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

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

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

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

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

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

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

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

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. The containment atmosphere particulate radioactivity monitoring system,

b. The containment sump level monitoring system, and

c. Either the containment fan cooler condensate flow rate or the containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) monitoring systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

SOURCE CHECK,

b. Containment sump level and containment fan cooler condensate flow rate (if being used) monitoring systems-performance of CHANNEL CALIBRATION at least once per 18 months.

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES LIMITING CONDITION FOR OPERATION

3.4.6.3 Reactor Coolant System Pressure Isolation Valves shall be operational.

a. The integrity of all pressure isolation valves listed in Table 4.4-~~1~~3 shall have been demonstrated, except as specified in "b". Valve leakage shall not exceed the amounts indicated.

b. In the event that the integrity of any pressure isolation valve specified in Table 4.4-~~1~~3 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain in, the mode corresponding to the ...lated condition.[a]

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

ACTION: If neither Condition "a" nor "b" can be met, an orderly shutdown shall be initiated within one hour and the reactor shall be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.3 a. Periodic leakage testing[b] on each valve listed in Table 4.4-~~1~~3 shall be accomplished:

1. Each time the plant is placed in COLD SHUTDOWN condition for refueling.

2. Each time the plant is placed in COLD SHUTDOWN condition for 72 hours if testing has not been accomplished in the preceding 12 months.

(a) Motor operated valves shall be placed in the closed position and power supplies deenergized.

(b) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

3. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve.

4. The provision of specification 4.0.4 is not applicable for entry into Mode 3 or 4.

b. Whenever integrity of a pressure isolation valve listed in Table 4.4-~~4~~3 cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure line shall be recorded daily.

TABLE ~~4.4-4~~ 3

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum(a)(b) Allowable Leakage
Low Pressure Safety Injection		≤5.0 GPM each valve
Loop 11, cold leg	11SJ56	≤5.0 GPM each valve
	11SJ43	≤5.0 GPM each valve
Loop 12, cold leg	12SJ56	≤5.0 GPM each valve
	12SJ43	≤5.0 GPM each valve
Loop 13, cold leg	13SJ56	≤5.0 GPM each valve
	13SJ43	≤5.0 GPM each valve
Loop 13, hot leg	13SJ156	≤5.0 GPM each valve
	13RH27	≤5.0 GPM each valve
Loop 14, cold leg	14SJ56	≤5.0 GPM each valve
	14SJ43	≤5.0 GPM each valve
Loop 14, hot leg	14SJ156	≤5.0 GPM each valve
	14RH27	≤5.0 GPM each valve
Intermediate Pressure Safety Injection		≤5.0 GPM each valve
Loop 11, cold leg	11SJ144	≤5.0 GPM each valve
Loop 11, hot leg	11SJ156	≤5.0 GPM each valve
	11SJ139	≤5.0 GPM each valve
Loop 12, cold leg	12SJ144	≤5.0 GPM each valve
Loop 12, hot leg	12SJ156	≤5.0 GPM each valve
	12SJ139	≤5.0 GPM each valve
Loop 13, cold leg	13SJ144	≤5.0 GPM each valve
Loop 13, hot leg	13SJ156	≤5.0 GPM each valve
	13SJ139	≤5.0 GPM each valve
Loop 14, cold leg	14SJ144	≤5.0 GPM each valve
Loop 14, hot leg	14SJ156	≤5.0 GPM each valve
	14SJ139	≤5.0 GPM each valve

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable. However, for initial tests, or tests following valve repair or replacement, leakage rates less than or equal to 5.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Minimum differential test pressure shall not be less than 150 psid.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq 100/\bar{E}\ \mu$Ci/gram.

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

ACTION:

MODES 1, 2 and 3*

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

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

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}\ \mu$Ci/gram, perform the sampling and analysis requirements of item 4a of Table ~~4.4-2~~ 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

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

*With T_{avg} $\geq$ 500°F.

OVERPRESSURE PROTECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each POPS shall be ~~demonstrated~~ OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the POPS actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the POPS is required OPERABLE, and at least once per 31 days thereafter when the POPS is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the POPS actuation channel at least once per 18 months.

c. Verifying the POPS ~~isolation valve~~ is open ~~at least~~ once per 72 hours when the POPS is being used for overpressure protection.

d. Testing pursuant to Specification 4.0.5.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vents(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.1.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated:

a. Per the requirements of Specification 4.0.5, and

b. Per the requirements of the augmented inservice inspection program specified in Specification 4.4.10.1.2.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - Tavg <350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump# and associated flow path capable of taking suction from the refueling water storage tank and transferring suction to the residual heat removal pump discharge piping and;

   1. Discharging into each Reactor Coolant System (RCS) cold leg.

b. One OPERABLE residual heat removal pump and associated residual heat removal heat exchanger and flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation and;

   1. Discharging into each RCS cold leg, and; upon manual initiation,

   2. Discharging into two RCS hot legs.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. Within no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

A maximum of one safety injection pump or one centrifugal charging pump shall be OPERABLE in MODE 4 when the temperature of one or more of the RCS cold legs is less than or equal to 312°F, Mode 5, or Mode 6 when the head is on the reactor vessel.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained volume of between 364,500 and 400,000 gallons of borated water.

b. A boron concentration of between 2,300 and 2,500 ppm, and

c. A minimum water temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the water level in the tank, and
2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is < 35°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

4.6.1.1 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 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 for valves that are opened under administrative control as permitted by Specification 3.6.3.1~~.~~, and

   2. All equipment hatches are closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

*Except vents, drains, test connections, etc. which are (1) one inch nominal pipe diameter or less, (2) located inside the containment, and (3) locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed at least once per 92 days.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of ≤ La, 0.10 percent by weight of the containment air per 24 hours at design pressure, (47.0 psig).

b. A combined leakage rate of ≤ 0.60 La for all penetrations and valves subject to Type B and C tests, when pressurized to Pa.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 La, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 La, 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 containment leakage rates shall be demonstrated as ~~at the~~ follows:

a. Type A tests shall be in accordance with 10CFR 50.54 (0) in conformance with Appendix J of 10CFR 50, Option B, using the methods and provisions of Regulatory Guide 1.163, September 1995 as modified by approved exemptions.

b. Type B and C tests shall be conducted in conformance with Appendix J of 10CFR 50, Option A, with gas at design pressure (47.0 psig) at intervals no greater than 24 months except for tests involving air locks.

c. Air locks shall be tested and demonstrated OPERABLE in conformance ~~convormance~~ with Appendix J of 10CFR 50, Option A, per surveillance Requirement 4.6.1.3.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test (reference Specification 4.6.1.2) to verify no apparent changes in appearance or other abnormal degradation. If the Type A test is performed at 10 year intervals, two additional inspections shall be performed at approximately equal intervals during shutdowns between Type A Tests.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be evaluated for reportability pursuant to 10CFR50.72 and 10CFR50.73. The evaluation shall be documented and shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective action taken.

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with:

a. A spray additive tank containing a volume of between 2568 and 4000 gallons of between 30 and 32 percent by weight NaOH solution, and

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the 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 spray additive system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

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

b. At least once per 6 months by:

1. Verifying the solution level in the tank, and

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

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 a Containment High-High pressure test signal.

d. At least once per 5 years by:

1. Verifying a NaOH solution flow rate of 12 ± 3 gpm from the spray additive tank through sample valve 1CS61 with the spray additive tank at 2.5 ± 0.5 psig and

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. Not used.

d. Verifying that on a Containment Purge and Pressure-Vacuum Relief isolation test signal, each Purge and Pressure-Vacuum Relief valve actuates to its isolation position.

e. Verifying that the Containment Pressure-Vacuum Relief Isolation valves are limited to ≤ 60° opening angle.

4.6.3.1.3 At least once per 18 months, verify that on a main steam isolation test signal, each main steam isolation valve actuates to its isolation position.

4.6.3.1.4 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.1.5 Each containment purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, then at least once per 72 hours, by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification ~~4.6.1.2d.~~ 4.6.1.2.b for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60La.

4.6.3.1.6 A pressure drop test to identify excessive degradation of resilient valve seals shall be conducted on the:

a. Containment Purge Supply and Exhaust Isolation Valves at least once per 6 months.

b. Deleted.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.~~l~~ m.

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

3/4.7.10 CHILLED WATER SYSTEM - AUXILIARY BUILDING SUBSYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10 The chilled water system loop which services the safety-related loads in the Auxiliary Building shall be OPERABLE with:

a. Three OPERABLE chillers

b. Two OPERABLE chilled water pumps

APPLICABILITY: ALL MODES and during movement of irradiated fuel assemblies.

ACTION: MODES 1, 2, 3, and 4

a. With one chiller inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 ~~hours and~~;

2. Restore the chiller to ~~operable~~ OPERABLE status within 14 days or;

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

b. With two chillers inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Align the control room emergency air conditioning system (CREACs) for single filtration operation using the Salem Unit 2 train within 4 hours and;

3. Restore at least one chiller to ~~operable~~ OPERABLE status within 72 hours or;

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

c. With one chilled water pump inoperable, restore the chilled water pump to ~~operable~~ 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.

PLANT SYSTEMS

LIMITING CONDITION FOR OPERATION

ACTION: MODES 5 and 6 or during movement of irradiated fuel assemblies. *

a. With one chiller inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Restore the chiller ~~to operable~~ OPERABLE status within 14 days or;

3. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

b. With two chillers inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Align the control room emergency air conditioning system (CREACs) for single filtration operation using the Salem Unit 2 train within 4 hours and;

3. Restore at least one chiller ~~to operable~~ OPERABLE status within 72 hours or;

4. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

c. With one chilled water pump inoperable, restore the chilled water pump ~~to operable~~ OPERABLE status within 7 days or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.10 The chilled water loop which services the safety-related loads in the Auxiliary Building shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each manual valve in the chilled water system flow path servicing safety related components that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months, by verifying that each automatic valve actuates to its correct position on a Safeguards Initiation signal.

c. At least once per 92 days by verifying that each chiller starts and runs.

* During Modes 5 and 6 and during movement of irradiated fuel assemblies, chilled water components are not considered to be inoperable solely on the basis that the backup emergency power source, diesel generator, is inoperable.

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. bus trains shall be OPERABLE and energized:

TRAIN 1A consisting of 125-volt D.C. bus No. 1A, 125-volt D.C. battery No. 1A and battery charger 1A1.

TRAIN 1B consisting of 125-volt D.C. bus No. 1B, 125-volt D.C. battery No. 1B and battery charger 1B1.

TRAIN 1C consisting of 125-volt D.C. bus No. 1C, 125-volt D.C. battery No. 1C and battery charger 1C1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one 125-volt D.C. bus inoperable or not energized, restore the inoperable bus to OPERABLE and energized status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 125-volt D.C. battery charger inoperable, restore the inoperable charger to OPERABLE status within 2 hours or connect the backup charger for no more than 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one or more 125-volt D.C. batteries with one or more battery cell parameters not within the Category A or B limits of Table 4.8.2.3-1:

   1. Verify within 1 hour, that the electrolyte level and float voltage for the pilot cell meets Table 4.8.2.3-1 Category C limits, and

   2. Verify within 24 hours, that the battery cell parameters of all connected cells meet Table 4.8.2.3-1 Category C limits, and

   3. Restore battery cell parameters to Category A and B limits of Table 4.8.2.3-1 within 31 days, and

   4. If any of the above listed requirements cannot be met, comply with the requirements of action f.

d. With one or more 125-volt D.C. batteries with one or more battery cell parameters not within Table 4.8.2.3-1 Category C values, comply with the requirements of action f.

e. With average electrolyte temperature of representative cells less than 65°F, comply with the requirements of action f.

f. Restore the battery to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

CRANE TRAVEL - FUEL HANDLING AREA

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2200 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage 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 overload cutoff which prevents crane travel with loads in excess of 2200 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least ~~one~~ once per 7 days thereafter during the crane operation.

Reference Change "c"

~~REFUELING OPERATIONS~~

~~COOLANT CIRCULATION~~

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

ALL WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.8.1 ~~At least one residual heat~~ removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual 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 containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual 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 REQUIREMETS

4.9.8.1 At least once per 12 hours one RHR loop shall be verified in operation and circulating coolant at a flow rate of:

a. greater than or equal to 1000 gpm, and

b. sufficient to maintain the RCS temperature at less than or equal to $140^{\circ}F$.

Reference change "z"

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops operable, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not ~~to~~ applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

---

* Systems supporting RHR loop operability may be excepted as follows:

a. The normal or emergency power source may be inoperable.

b. One service water header may be out of service provided the equipment listed in Table 3.4-3 is OPERABLE.

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

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 operations involving movement of fuel assemblies or control rods within the 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 thereafter during movements of fuel assemblies or control rods.

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s). ~~and~~

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length rod either partially or FULLY WITHDRAWN shall be determined at least once per 2 hours.

4.10.1.2 Each full length 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.

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined in Table ~~4.11.1~~ 4.11-1.

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

c. Sampling and analysis shall also be performed following shutdown, startup or a THERMAL POWER change that, within one hour, exceeds 15 percent of RATED THERMAL POWER unless:

1. Analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of three.

2. The noble gas activity monitor shows that effluent activity has not increased by more than a factor of three.

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

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

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

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 Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. 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 that specific site parameters of the Salem 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.

Reducing T_{avg} to $<500°F$ prevents the release of activity should a steam generator tube rupture occur 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

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rate (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of non ductile ~~nonductile~~ failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods provided below.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975".

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 15 effective full power years of service life. The 15 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

3/4.7.9 SNUBBERS

All snubbers 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. Snubbers 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 were installed, would have no adverse effect on any safety related system.

A list of individual snubbers required to be operable per the technical specifications with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Station Operations Review Committee. The determination shall be based on the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.) and the recommendations of Regulatory Guide 8.8 and 8.10. The addition or deletion of any snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. The inspections are performed for each category of snubbers. The snubbers are categorized by accessibility (i.e., accessible or inaccessible during reactor operation). The next visual inspection for each category may be twice, the same, or reduced by as much as two-thirds of the previous inspection interval. This interval depends on the number of unacceptable snubbers found in proportion to the total number of snubbers in each category from the most recent inspection. Intervals may be increased up to 48 months if few unacceptable snubbers are found in these inspections. The visual inspection interval will not exceed 48 months. However, as for all surveillance activities, unless otherwise noted, allowable tolerances of 25% are applicable for snubbers. Table 4.7-3 establishes three limits for determining the next visual inspection interval corresponding to the population of each category of snubbers. For a category that differs from the representative sizes provided, the values for the next inspection interval may be found by interpolation from the limits provided in Columns A, B, and C. Where the limit for unacceptable snubbers in Columns A, B, or C is determined by interpolation and includes a fractional value, the limit may be reduced to the next lower integer. The first inspection interval determined using Table 4.7-3 shall be based upon the previous inspection interval as established by the requirements in effect before amendment (161). Any inspection whose results require a shorter inspection interval will override the previous schedule.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 47 psig. Containment air temperatures up to 351.3°F are acceptable providing the containment pressure is in accordance with that described in the UFSAR.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. 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 ~~FEATURE~~ AND TEMPERATURE

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

such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be ~~reasonable~~ reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or ~~remote (such~~ as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection ~~procedures to~~ provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SORC.

2. Shall become effective upon review and acceptance by the SORC.

Unit 2

Mark-Up Pages

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ADMINISTRATIVE CONTROLS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

1.7.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are opened under administrative control as permitted by Specification 3.6.3.1.

1.7.2 All equipment hatches are closed and sealed,

1.7.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.7.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.7.5 The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

1.8 NOT USED

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 9.0% of narrow range instrument span-each steam generator	≥ 8.0% of narrow range instrument span-each steam generator
14. Deleted		
15. Undervoltage-Reactor Coolant Pumps	≥ 2900 volts-each bus	≥ 2850 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 56.5 Hz - each bus	≥ 56.4 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	≥ 45 psig	≥ 45 psig
B. Turbine Stop Valve Closure	≤ 15% off full open	≤ 15% off full open
18. Safety Injection Input from ~~SSPS~~ ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition ~~shall:~~

(a) shall not be made when the conditions of the Limiting Condition for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval.

(b) may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time.

This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.6% delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.6% delta k/k, immediately initiate and continue boration at ≥ 33 gpm of a solution containing ≥ 6,560 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.6% delta k/k:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control ~~bank withdrawal is~~ banks are within the limits of Specification 3.1.3.5.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.5.

*See Special Test Exception 3.10.1

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0 delta k/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition.

b. Less negative than -4.4×10^{-4} delta k/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3.a - MODES 1 and 2* only#
Specification 3.1.1.3.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.3.a, above, operations in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 delta k/k/°F within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification ~~3.1.3.6.~~ 3.1.3.5.

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. In lieu of any other report required by Specification 6.9.1, a Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.3.b, above, be in HOT SHUTDOWN within 12 hours.

*With Keff greater than or equal to 1.0

#See Special Test Exception 3.10.3

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods, shall be OPERABLE and positioned within ± 18 steps (indicated position) when reactor power is ≤ 85% RATED THERMAL POWER, or ± 12 steps (indicated position) when reactor power is > 85% RATED THERMAL POWER, of their group step counter demand position within one hour after rod motion.

APPLICABILITY: MODES 1* and 2*

ACTION:

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

b. With more than one full length rod inoperable or mis-aligned from the group step counter demand position by more than ± 18 steps (indicated position) at ≤ 85% RATED THERMAL POWER or ± 12 steps (indicated position) at > 85% RATED THERMAL POWER, be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or mis-aligned from its group step counter demand position by more than ± 18 steps (indicated position) at ≤ 85% RATED THERMAL POWER or ± 12 steps (indicated position) at > 85% RATED THERMAL POWER, POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The remainder of the rods in the bank with the inoperable rod are ~~aligned to~~ within ± 18 steps (indicated position) at ≤ 85% RATED THERMAL POWER or ± 12 steps (indicated position) at >85% RATED THERMAL POWER, of the inoperable rod while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-23; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

*See Special Test Exceptions 3.10.2 and 3.10.3.

~~POSITION INDICATION SYSTEM SHUTDOWN~~ SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be FULLY WITHDRAWN.

APPLICABILITY: MODES 1*, and 2*#@.

ACTION:

With a maximum of one shutdown rod not FULLY WITHDRAWN, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. FULLY WITHDRAW the rod, or,

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be FULLY WITHDRAWN by use of the group demand counters, and verified by the analog rod position indicators**:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, and D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

* See Special Test Exceptions 3.10.2 and 3.10.3.
** For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.
@ Surveillance 4.1.3.4.a is applicable prior to withdrawing any control banks in preparation for startup (Mode 2).
\# With Keff greater than or equal to 1.0.

Note: This page effective prior to startup from fifth refueling outage scheduled to begin March 1990. Letter dated Jan. 11, 1990.

REACTIVITY CONTROL SYSTEMS

~~POSITION INDICATION SYSTEM SHUTDOWN~~ CONTROL ROD INSERTION LIMITS

~~LIMITING CONDITION FOR OPERATION~~

3.1.3.5 The control banks shall be limited in physical insertion as shown in Figures 3.1-1 and ~~3.1-2.~~ 3.1-3.

APPLICABILITY: ~~MODES 1*, and 2*#~~

Change "nn"

Change "b"

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the control banks to within the limits within two hours, or

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figures, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours by use of the group demand counters and verified by the analog rod position indicators** except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours**.

*See Special Test Exceptions 3.10.2 and 3.10.3
**For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.
#With Keff greater than or equal to 1.0

FIGURE 3.1-~~2~~3

INTENTIONALLY LEFT BLANK PENDING
COMMISSION APPROVAL OF THREE LOOP OPERATION

FIGURE 3.1-~~2~~3 ROD BANK INSERTION LIMITS VERSUS THERMAL
POWER FOR THREE LOOP OPERATION

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the +6, -9% target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the +6, -9% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours. Power increases above 50% of RATED THERMAL POWER do not require being within the target band provided the accumulative penalty deviation is not violated.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its +6, -9% target band when at least 2 or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the +6, -9% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels below 50% of RATED THERMAL POWER.

TABLE 3.3-1 (Continued)

ACTION 10 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY in the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1 provided the other channel is OPERABLE.

ACTION 11 - With less than the Minimum Number of Channels OPERABLE, operation may continue provided the inoperable channel is placed in the tripped condition within 6 hours.

ACTION 12 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 13 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 14 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and be in at least HOT STANDBY within 6 hours. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels < 6×10^{-11} amps.	P-6 prevents or defeats the manual block of source range reactor trip.
P-7	With 2 of 4 Power Range Neutron Flux Channels ≥ 11% of RATED THERMAL POWER or 1 of 2 Turbine Impulse chamber pressure channels ≥ a pressure equivalent to 11% of RATED THERMAL POWER.	P-7 prevents or defeats ~~Flux~~ the automatic block of reactor trip on: Low flow in more than one primary coolant loop, reactor coolant pump undervoltage and under-frequency, pressurizer low pressure, pressurizer high level, and the opening of more than one reactor coolant pump breaker.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	R[9]	1, 2, and *
2. Power Range, Neutron Flux	S	D[2], M[3] and Q[6]	Q	1, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R[6]	Q	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R[6]	Q	1, 2
5. Intermediate Range, Neutron Flux	S	R[6]	S/U[1]	1, 2 and *
6. Source Range, Neutron Flux	S[7]	R[6]	Q and S/U[1]	2, 3, 4, 5 and *
7. Overtemperature ΔT	S	R	Q	1, 2
8. Overpower ΔT	S	R	Q	1, 2
9. Pressurizer Pressure--Low	S	R	Q	1, 2
10. Pressurizer Pressure--High	S	R	Q	1, 2
11. Pressurizer Water Level--High	S	R	Q	1, 2
12. Loss of Flow - Single Loop	S	R	Q	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow Two Loops	S	R	N.A.	1
14. Steam Generator Water Level--Low-Low	S	R	Q	1, 2
15. Deleted				
16. Undervoltage - Reactor Coolant Pumps	N.A.	R	Q	1
17. Underfrequency - Reactor Coolant Pumps	N.A.	R	Q	1
18. Turbine Trip				
a. Low Autostop Oil Pressure	N.A.	N.A.	S/U(1)	~~N.A.~~ 1, 2
b. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	~~N.A.~~ 1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)(5)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	~~N.A.~~ 1
21. Reactor Trip Breaker	N.A.	N.A.	M(5)(11)(13) and R(14)	1, 2 and *
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2 and *

Change "gg"

Change ";"

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

SOURCE CHECK,

TABLE 3.[illegible] (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	$\leq 3.0 \times 10^{-2} \mu Ci/cm^3$ (Alarm only)	10^{-3}-10^{1} $\mu Ci/cm^3$	26
2) High Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	$\leq 1.0 \times 10^{2} \mu Ci/cm^3$ (Alarm only)	10^{-1}-10^{5} $\mu Ci/cm^3$	26
3) Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps)	1/ MS Line	1,2,3&4	10 mR/hr (Alarm only)	1-10^{4} mR/hr	26
4) Condenser Exhaust System	1	1,2,3&4	$\leq 7.12 \times 10^{4}$ cpm (Alarm only)	1-10^{6} cpm	26
3. CONTROL ROOM					
a. Air Intake - Radiation Level	2/Intake##	**	$\leq 2.48 \times 10^{3}$ cpm	10^{1}-10^{7} cpm	27, 28

Change "cc"

Control Room air intakes shared between Unit 1 and 2.

** ALL MODES and during movement of irradiated fuel assemblies and during ~~core alterations~~ CORE ALTERATIONS.

Change "uu"

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.7.1.

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 26 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

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

ACTION 27 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel(s) to OPERABLE status within 7 days or initiate and maintain operation of the Control Room Emergency Air Conditioning System (CREACS) in the pressurization or recirculation mode of operation. CORE ALTERATIONS and movement of irradiated fuel assemblies will be suspended during operation in the recirculation mode.

ACTION 28 With no channels OPERABLE in a Control Room air intake, immediately initiate and maintain operation of the CREACS in the pressurization or recirculation mode of operation. CORE ALTERATIONS and movement of irradiated fuel assemblies will be suspended during operation in the recirculation mode.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be operable.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. As shown in Table 3.3-11.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, ~~AND~~ CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-11.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1	1, 2
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1	1, 2
3. Reactor Coolant Pressure (Wide Range)	2	1	1, 2
4. Pressurizer Water Level	2	1	1, 2
5. Steam Line Pressure	2/Steam Generator	1/Steam Generator	1, 2
6. Steam Generator Water Level (Narrow Range)	2/Steam Generator	1/Steam Generator	1, 2
7. Steam Generator Water Level (Wide Range)	4 (1/Steam Generator)	3 (1/Steam Generator)	1, 2
8. Refueling Water Storage Tank Water Level	2	1	1, 2
9. deleted			
10. Auxiliary Feedwater Flow Rate	4 (1/Steam Generator)	3 (1/Steam Generator)	4, 6
11. Reactor Coolant System Subcooling Margin Monitor	2	1	1, 2
12. PORV Position Indicator	2/valve**	1	1, 2

TABLE 3.3-11 (continued)

TABLE NOTATION

ACTION 1 With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 2 With the number of OPERABLE accident monitoring channels less than the Minimum Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in HOT SHUTDOWN within the next 12 hours.

ACTION 3 deleted

ACTION 4 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operations may proceed provided that an OPERABLE Steam Generator Wide Range Level channel is available as an alternate means of indication for the Steam Generator with no OPERABLE Auxiliary Feedwater Flow Rate Channel.

ACTION 5 ~~With the number of OPERABLE channels less than the Required Number of Channels shown in Table 3.3-11, operation may proceed provided that Steam Tables are available in the Control Room and the following Required Channels shown in Table 3.3-11 are OPERABLE to provide an alternate means of calculating Reactor Coolant System subcooling margin:~~

~~a. Reactor Coolant Outlet Temperature - T (Wide Range)~~

~~b. Reactor Coolant Pressure (Wide Range)~~

deleted

TABLE 4.3-11
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL ~~TEST~~
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R	~~NA~~ N.A.
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R	~~NA~~ N.A.
3. Reactor Coolant Pressure (Wide Range)	M	R	~~NA~~ N.A.
4. Pressurizer Water Level	M	R	~~NA~~ N.A.
5. Steam Line Pressure	M	R	~~NA~~ N.A.
6. Steam Generator Water Level (Narrow Range)	M	R	~~NA~~ N.A.
7. Steam Generator Water Level (Wide Range)	M	R	~~NA~~ N.A.
8. Refueling Water Storage Tank Water Level	M	R	~~NA~~ N.A.
9. deleted			
10. Auxiliary Feedwater Flow Rate	~~SU#~~ S/U#	R	~~NA~~ N.A.
11. Reactor Coolant System Subcooling Margin Monitor	M	~~N/A*~~ N.A.*	~~NA~~ N.A.

Change "hh"

Change "u"

#Auxiliary Feedwater System is used on each startup and flow rate indication is verified at that time.

*The instruments used to develop RCS subcooling margin are calibrated on an 18 month cycle; the monitor will be compared quarterly with calculated subcooling margin for known input values.

TABLE 4.3-11 (Continued)
SURVEILLANCE REQUIREMENTS FOR ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
12. PORV Position Indicator	M	~~NA~~ N.A.	R
13. PORV Block Valve Position Indicator	M	~~NA~~ N.A.	Q*
14. Pressurizer Safety Valve Position Indicator	M	~~NA~~ N.A.	R
15. Containment Pressure - Narrow Range	M	~~NA~~ R	~~NA~~ N.A.
16. Containment Pressure - Wide Range	M	R	~~NA~~ N.A.
17. Containment Water Level - Wide Range	M	R	~~NA~~ N.A.
18. Core Exit Thermocouples	M	R	~~NA~~ N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)	M	R	~~NA~~ N.A.

Change "bb"

Change "hh"

* Unless the block valve is closed in order to meet the requirements of Action b, or c in specification 3.4.5.

TABLE 4.3-12 (Continued)

TABLE NOTATION

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

1. Instrument indicates measured levels at or above the alarm/trip setpoint.
2. Circuit failure. (Loss of Power) (Automatic Isolation only)
3. Instrument indicates a downscale failure.

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

1. Instrument indicates measured levels at or above the alarm/trip setpoint.
2. Circuit failure. (Loss of Power) (Indication only)
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION was performed using appropriate liquid or gaseous calibration sources obtained from reputable suppliers. The activity of the calibration sources were reconfirmed using a multi-channel analyzer which was calibrated using one or more NBS standards.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

* During liquid additions to the tank.

** If tank level indication is not provided, vertification will be done by visual inspection.

\# The R18 channel is an off-line channel which requires periodic decontamination. Any count rate indication above 10,000 cpm constitutes a SOURCE ~~CHANNEL~~ CHECK for compliance purposes.

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL ~~OPERABLE~~ CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	*
b. Oxygen Monitor	D	N.A.	Q(4)	M	**
2. CONTAINMENT PURGE AND PRESSURE - VACUUM RELIEF					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	***
3. PLANT VENT HEADER SYSTEM#					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	N.A.	*
e. Sampler Flow Rate Monitor	W	N.A.	R	N.A.	*

\# The following process streams are routed to the plant vent where they are effectively monitored by the instruments described:

(a) Condenser Air Removal System
(b) Auxiliary Building Ventilation System
(c) Fuel Handling Building Ventilation System
(d) Radwaste Area Ventilation System
(e) Containment Purges

SURVEILLANCE REQUIREMENTS (Continued)

4.3.4.3 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

b. At least ~~one~~ once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

4.3.4.4 Verify the test frequency maintains the probability of a missile ejection incident within NRC guidelines by reviewing the methodology presented in WCAP-11525:

a. At least once every two refueling outages.

b. After modifications to the main turbine or turbine overspeed protection valves.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 21 and its associated steam generator and reactor coolant pump,*
2. Reactor Coolant Loop 22 and its associated steam generator and reactor coolant pump,*
3. Reactor Coolant Loop 23 and its associated steam generator and reactor coolant pump,*
4. Reactor Coolant Loop 24 and its associated steam generator and reactor coolant pump,*
5. Residual Heat Removal Loop 21,
6. Residual Heat Removal Loop 22.

b. At least one of the above coolant loops shall be in operation.**

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

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

*A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 312°F unless 1) the pressurizer water volume is less than 1650 cubic feet (equivalent to approximately 92% of level) or 2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

**All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4.7 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.7.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. The containment atmosphere particulate radioactivity monitoring system,

b. The containment pocket sump level monitoring system, and

c. Either the containment fan cooler condensate flow rate or the containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity monitoring system is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere particulate and gaseous (if being used) SOURCE CHECK monitoring systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3.

b. Containment pocket sump level and containment fan cooler condensate flow rate (if being used) monitoring systems-performance of CHANNEL CALIBRATION at least once per 18 months.

3/4.4.9 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq 100/\bar{E}$ μCi/gram.

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

ACTION:

MODES 1, 2 and 3*

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

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

MODES 1, 2, 3, 4 and 5

a. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}$ μCi/gram, perform the sampling and analysis requirements of item 4a of Table ~~4.4-2~~ 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

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

*With $T_{avg} \geq$ 500°F.

REACTOR COOLANT SYSTEM

3.4.11 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.11.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification ~~4.4.10.1~~. 4.4.11.1

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11.1 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

4.4.11.2 Augmented Inservice Inspection Program for Steam Generator Channel Heads - The No. 21 Steam Generator channel head shall be ultrasonically inspected in a selected area during each of the first three refueling outages using the same ultrasonic inspection procedures and equipment used to generate the baseline data. These inservice ultrasonic inspections shall verify that the cracks observed in the stainless steel cladding prior to operation have not propagated into the base material.

ECCS SUBSYSTEMS - T_{avg} <350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump# and associated flow path capable of taking suction from the refueling water storage tank and transferring suction to the residual heat removal pump discharge piping and;

   1. Discharging into each Reactor Coolant System (RCS) cold leg.

b. One OPERABLE residual heat removal pump and associated residual heat removal heat exchanger and flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation and;

   1. Discharging into each RCS cold leg, and; upon manual initiation,

   2. Discharging into two RCS hot legs.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With~~in~~ no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or residual heat removal pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

A maximum of one safety injection pump or one centrifugal charging pump shall be OPERABLE in MODE 4 when the temperature of one or more of the RCS cold legs is less than or equal to 312°F, Mode 5, or Mode 6 when the head is on the reactor vessel.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A contained volume of between 364,500 and 400,000 gallons of borated water.

b. A boron concentration of between 2,300 and 2,500 ppm, and

c. A minimum water temperature of 35°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is < 35°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

Change "ccc"

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE 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 for valves that are ~~may be~~ opened under Administrative control as permitted by Specification 3.6.3.~~1~~ e,
and all equipment hatches are closed and sealed.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

Change "e"

c. After each closing of a penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at Pa ~~(47 psig)~~ and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.~~d~~ be for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

Change "g"

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* Except vents, drains, test connections, etc. which are (1) one inch nominal pipe diameter or less, (2) located inside the containment, and (3) locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed at least once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to La, 0.10 percent by weight of the containment air per 24 hours at design pressure (47.0 psig).

b. A combined leakage rate of less than or equal to 0.60 La for all penetrations and valves subject to Type B and C tests ~~as identified in Table 3.6-1,~~ when pressurized to Pa.

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

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 La, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 La, 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 containment leakage rates shall be demonstrated as follows:

a. Type A tests shall be in accordance with 10CFR 50.54(0) in conformance with Appendix J of 10CFR 50, Option B, using the methods and provisions of Regulatory Guide 1.163, September 1995 as modified by approved exemptions.

b. Type B and C tests shall be conducted in conformance with Appendix J of 10CFR 50, Option A, with gas at design pressure (47.0 psig) at intervals no greater than 24 months except for tests involving air locks.

c. Air locks shall be tested and demonstrated OPERABLE in conformance with Appendix J of 10CFR 50, Option A, per Surveillance Requirement 4.6.1.3.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the 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

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined by a visual inspection of these surfaces. The inspection shall be performed prior to the Type A containment leakage rate test (reference Specification 4.6.1.2) to verify no apparent changes in appearance or other abnormal degradation. If the Type A test is performed at 10 year intervals, two additional inspections shall be performed at approximately equal intervals during shutdowns between Type A tests.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required inspections shall be evaluated for reportability pursuant to 10CFR50.72 and 10CFR50.73. The evaluation shall be documented and shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective action taken.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST and transferring suction to the RHR pump discharge.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one 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 following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 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, is in its correct position.

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 215 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 a Containment High-High pressure test signal.

   2. Verifying each ~~each~~ a spray pump starts automatically on a Containment High-High pressure test signal.

d. At least once per 10 years by:

   1. Performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase A containment isolation test signal, each Phase A isolation valve actuates to its isolation position.

b. Verifying that on a Phase B containment isolation test signal, each Phase B isolation valve actuates to its isolation position.

c. NOT USED

d. Verifying that on a Containment Purge and Pressure-Vacuum Relief isolation test signal, each Purge and Pressure-Vacuum Relief valve actuates to its isolation position.

e. Verifying that the Containment Pressure-Vacuum Relief Isolation valves are limited to ≤ 60° opening angle.

4.6.3.3 At least once per 18 months, verify that on a main steam isolation test signal, each main steam isolation valve actuates to its isolation position.

4.6.3.4 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.5 Each containment purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, ~~then~~ at least once per 72 hours, by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification ~~4.6.1.2d~~ 4.6.1.2.b for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60La.

4.6.3.6 A pressure drop test to identify excessive degradation of resilient valve seals shall be conducted on the:

a. Containment Purge Supply and Exhaust Isolation Valves at least once per 6 months.

b. Deleted.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODES 1 - With one main steam 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 MODE 2 within the next 6 hours.

MODES 2 and 3 - With one or more main steam line isolation valve(s) inoperable, subsequent operation in MODES 2 or 3 may proceed provided;

a. The isolation valve(s) is (are) maintained closed, and

b. The isolation valve(s) is (are) verified closed once per 7 days.

Otherwise, be in MODE 3, HOT STANDBY, within the next 6 hours, and MODE 4, HOT SHUTDOWN, within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of ~~Specificaiton~~ Specification 4.0.4 are not applicable.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.~~l~~. M

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

PLANT SYSTEMS

3/4.7.10 CHILLED WATER SYSTEM - AUXILIARY BUILDING SUBSYSTEM

LIMITING CONDITION FOR OPERATION

3.7.10 The chilled water system loop which services the safety-related loads in the Auxiliary Building shall be OPERABLE with:

a. Three OPERABLE chillers

b. Two OPERABLE chilled water pumps

APPLICABILITY: ALL MODES and during movement of irradiated fuel assemblies.

ACTION: MODES 1, 2, 3, and 4

a. With one chiller inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Restore the chiller to OPERABLE ~~operable~~ status within 14 days or;

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

b. With two chillers inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Align the control room emergency air conditioning system (CREACs) for single filtration operation using the Salem Unit 1 train within 4 hours and;

3. Restore at least one chiller to OPERABLE ~~operable~~ status within 72 hours or;

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

c. With one chilled water pump inoperable, restore the chilled water pump to OPERABLE ~~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

LIMITING CONDITION FOR OPERATION

ACTION: MODES 5 and 6 or during movement of irradiated fuel assemblies.*

a. With one chiller inoperable:

1. Remove the appropriate non-essential heat loads from the Chilled Water System within 4 hours and;

2. Restore the chiller to ~~operable~~ OPERABLE status within 14 days or;

3. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

b. With two chillers inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Align the control room emergency air conditioning system (CREACs) for single filtration operation using the Salem Unit 1 train within 4 hours and;

3. Restore at least one chiller to ~~operable~~ OPERABLE status within 72 hours or;

4. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

c. With one chilled water pump inoperable, restore the chilled water pump to ~~operable~~ OPERABLE status within 7 days or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.10 The chilled water loop which services the safety-related loads in the Auxiliary Building shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each manual valve in the chilled water system flow path servicing safety related components that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months, by verifying that each automatic valve actuates to its correct position on a Safeguards Initiation signal.

c. At least once per 92 days by verifying that each chillers starts and runs.

* During Modes 5 and 6 and during movement of irradiated fuel assemblies, chilled water components are not considered to be inoperable solely on the basis that the backup emergency power source, diesel generator, is inoperable.

ELECTRICAL POWER SYSTEMS

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 from sources of power other than the diesel generators:

4 kvolt	Vital Bus # 2A	
4 kvolt	Vital Bus # 2B	
4 kvolt	Vital Bus # 2C	
460 volt	Vital Bus # 2A	and associated control centers
460 volt	Vital Bus # 2B	and associated control centers
460 volt	Vital Bus # 2C	and associated control centers
230 volt	Vital Bus # 2A	and associated control centers
230 volt	Vital Bus # 2B	and associated control centers
230 volt	Vital Bus # 2C	and associated control centers
115 volt	Vital Instrument Bus # 2A	and Inverter*
115 volt	Vital Instrument Bus # 2B	and Inverter*
115 volt	Vital Instrument Bus # 2C	and Inverter*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With less than the above complement of A.C. busses OPERABLE or energized, restore the inoperable busses to OPERABLE and energized 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.

b. With one inverter inoperable, energize the associated A.C. Vital Bus within 8 hours; restore the inoperable inverter to OPERABLE and energized status 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

4.8.2.1 The specified A.C. busses and inverters shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

*An inverter may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided (1) its vital bus is OPERABLE and energized, and (2) the vital busses associated with the other battery banks are OPERABLE and energized.

SURVEILLANCE REQUIREMENTS (Continued)

2. The pilot cell specific gravity, corrected to 77°F, and full electrolyte level, is greater than or equal to 1.200.

3. The pilot cell voltage is greater than or equal to 2.08 volts.

4. The overall battery voltage is greater than or equal to 27 volts.

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

1. The voltage of each connected cell is greater than or equal to 2.13 volts under float charge and has not decreased more than 0.27 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 greater than or equal to 1.200 and has not decreased more than 0.02 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 abnormal 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 150 amperes at 28 volts for at least 4 hours.

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 or simulated emergency loads for the design duty cycle 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. Satisfactory completion of this performance discharge test shall also satisfy the requirements of Specification 4.8.2.5.2.d if the performance discharge test is conducted during a shutdown where that test and the battery service test would both be required.

REFUELING OPERATIONS

CRANE TRAVEL - FUEL HANDLING AREA

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2200 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage 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 overload cutoff which prevents crane travel with loads in excess of 2200 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least ~~one~~ once per 7 days thereafter during the crane operation.

Reference Change "C"

3/4.9.8 RESIDUAL HEAT REMOVAL and COOLANT CIRCULATION

ALL WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one residual 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 containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The residual 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

4.9.8.1 At least once per 12 hours one RHR loop shall be verified in operation and circulating coolant at a flow rate of:

a. greater than or equal to 1000 gpm, and

b. sufficient to maintain the RCS temperature at less than or equal to 140°F.

Reference change "Z"

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops operable, immediately initiate corrective action to return ~~the~~ required RHR loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are ~~to~~ not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8.2 The required Residual Heat Removal loops shall be determined OPERABLE per Specification 4.0.5.

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* Systems supporting RHR loop operability may be excepted as follows:

a. The normal or emergency power source may be inoperable.

b. One service water header may be out of service provided the equipment listed in Table 3.4-3 is OPERABLE.

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined in Table ~~4.11.1~~ 4.11-1

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

c. Sampling and analysis shall also be performed following shutdown, startup or a THERMAL POWER change that, within one hour, exceeds 15 percent of RATED THERMAL POWER unless:

   1. Analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of three.

   2. The noble gas activity monitor shows that effluent activity has not increased by more than a factor of three.

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

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

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

3/4.4.9 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. 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 that specific site parameters of the Salem 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.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur 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.

BASES

Finally, the new 10CFR50 rule which addresses the metal temperature of the closure head flange regions is considered. This 10CFR50 rule states that the metal temperature of the closure flange regions must exceed the material RTNDT by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Salem). Table B3/4.4-1 indicates that the limiting RT_{NDT} of 28°F occurs in the closure head flange of Salem Unit 1, and the minimum allowable temperature of this region is 148°F at pressures greater than 621 psig. These limits do not affect Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two POPSs or an RCS vent opening of greater than 3.14 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold ~~leg~~ legs are less than or equal to 312°F. Either POPS has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of an Intermediate Head Safety Injection pump pump and its injection into a water solid RCS, or the start of a High Head Safety Injection pump in conjunction with a running Positive Displacement pump and its injection into a water solid RCS.

BASES

3/4.10.1 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.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth, and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the Reactor Coolant System T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the reactor core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is, at times, necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not be allowed by Specification ~~3.1.3.6~~ which may, in turn, cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3.1.3.5

3/4.10.4 NO FLOW TESTS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment is designed and shall be maintained for a maximum internal pressure of 47 psig. Containment air temperatures up to 351.3°F are acceptable providing the containment pressure is in accordance with that described in the UFSAR.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of zircaloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. 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 ~~FEATURE~~ PRESSURE AND TEMPERATURE

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

a. In accordance with the code requirement specified in Section 4.1 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,811 ± 100 cubic feet at a nominal T_{avg} of 581.0°F.