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June 8, 1984

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

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: Catawba Nuclear Station, Unit 1
Docket No. 50-413
Draft Technical Specifications

Dear Mr. Denton:

Please find attached proposed changes to the Draft Technical Specifications for Catawba Unit 1. These changes supply missing information and make corrections to errors presently contained in the Specifications.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

RWO/php

Attachment

cc: Mr. James P. O'Reilly
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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SECTION PAGE

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for four loop operation~~, respectively~~.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤111.1% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤31% of RTP*
6. Source Range, Neutron Flux	17.0	10	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	7.2	4.47	2.~~[illegible]~~ .03	See Note 1	See Note 2
8. Overpower ΔT	4.3	1.3	1.2	See Note 3	See Note 4
9. Pressurizer Pressure-Low	4.0	2.21	1.5	≥1945 psig	≥1938 psig***
10. Pressurizer Pressure-High	7.5	4.96	0~~1~~.5	≤2385 psig	≤2399 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.77	0.6	≥90% of loop design flow**	≥89.2% of loop design flow**

*RTP = RATED THERMAL POWER
**Loop design flow = 96,900 gpm
***Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to these values.

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low	17	14.2	1.5	≥17% of span from 0% to 30% RTP* increasing linearly to ≥ ~~59.9%~~ 54.9 of span from 30% to 100% RTP*	≥15.3% of span from 0% to 30% RTP* increasing linearly to ≥~~58.2%~~ 53.2 of span from 30% to 100% RTP*
14. Undervoltage - Reactor Coolant Pumps	8.57	0	1.0	≥77% of bus voltage (5082 volts) with a 0.7s response time	≥76% (5016 volts)
15. Underfrequency - Reactor Coolant Pumps	4.0	0	1.0	≥56.4 Hz with a 0.2s response time	≥55.9 Hz
16. Turbine Trip					
a. Low Control Valve EH Pressure	N.A.	N.A.	N.A.	≥550 psig	≥500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

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2.1 SAFETY LIMITS

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BASES

~~These~~ curve~~s~~ ~~are~~ based on a nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, ~~of~~ 1.49 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:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)]$$

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 $f_1\ (\Delta I)$ 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 vessel, pressurizer, and the Reactor Coolant System piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of 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 3110 ~~3107~~ psig, 125% of design pressure, to demonstrate integrity prior to ~~initial~~ operation.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor STARTUP to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature-induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.2-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature-induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors~~, and (3) axial power distribution,~~ to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

LIMITING SAFETY SYSTEM SETTINGS

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BASES

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, is automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, is automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection and ~~to~~ prevents DBN by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 48% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and ~~the~~ P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

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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 greater than or equal to 1.3% Δk/k for four loop operation.

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

With the SHUTDOWN MARGIN less than 1.3% Δk/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 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 1.3% Δk/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 verified acceptable with an increased allowance for 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 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. When in MODE 2 with K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. ~~Prior to~~ initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1e. below, with the control banks at the maximum inservice limit of Specification 3.1.3.6; and

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*See Special Test Exception Specification 3.10.1.

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MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

.1

3.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 551°F.

APPLICABILITY: MODES 1 and 2#*.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System Temperature (T_{avg}) shall be determined to be greater than or equal to 551°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

*With K_{eff} greater than or equal to 1.

#See Special Test Exception Specification 3.10.3.

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REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - ~~SHUTDOWN~~ OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two* charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying that a differential pressure across each pump of greater than or equal to 2390 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 31 days whenever the temperature of one of more of the Reactor Coolant System cold legs is less than or equal to 285°F by verifying that the motor circuit breakers are secured in the open position or that the discharge of each charging pump has been isolated from the Reactor Coolant System by at least two isolation valves with power removed from the valve motor operators.

*A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the Reactor Coolant System cold legs is less than or equal to 285°F.

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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.

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 misaligned 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 trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ±12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

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

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuance to Specification 3.1.3.6 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:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

      b) The SHUTDOWN MARGIN requirement~~s~~ of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exception~~s~~ Specifications 3.10.2 and 3.10.3.

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ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

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POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3*#, 4*#, and 5*#.

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicator agrees with the demand position indicator within 12 steps when exercised over the full-range of rod travel at least once per 18 months.

*With the Reactor Trip System breakers in the closed position.
#See Special Test Exception Specification 3.10.5

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CONTROL BANK INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figure 3.1-1.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figure, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours 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 Specifications 3.10.2 and 3.10.3.
#With K_{eff} greater than or equal to 1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

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LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. ± 5% for Cycle 1 core average accumulated burnup of less than or equal to 5000 MWD/MTU;

b. + 3%, -9% for Cycle 1 core average accumulated burnup of greater than 5000 MWD/MTU; and

c. +3%, -12% for subsequent cycles.

The indicated AFD may deviate outside the above required target level at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes, either:

1. Restore the indicated AFD to within the target band limits, or

2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time~~s~~ during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

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*See Special Test Exceptions Specification 3.10.2.

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. The Power Range Neutron Flux* - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD 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 AFD 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 AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

*Surveillance testing of the Power Range Neutron Flux Channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

3/4.2.4 QUADRANT POWER TILT RATIO

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LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02 above 50% of RATED THERMAL POWER.

APPLICABILITY: MODE 1.*

ACTION:

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

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

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

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

2. Within 2 hours either:

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

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

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

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

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*See Special Test Exception Specification 3.10.2.

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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

#The provisions of Specification 3.0.4 are not applicable.
##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.
####Above the P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

ACTION STATEMENTS

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

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

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

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

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

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

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

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE REQUIRED
18. Reactor Trip System Interlocks (Continued)						
e. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1, 2
f. Turbine Impulse Chamber Pressure, P-13	N.A.	R	M(8)	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

Above P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

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

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

(1) If not performed in previous 7 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable ~~to~~ for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore axial flux difference above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) With power greater than or equal to the interlock setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive status light.

(9) Monthly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive status light. Monthly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to one-half decade above background.

(10) Setpoint verification is not applicable.

(11) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

(12) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	12.7	0.71	1.5	$\leq$ 3 psig	$\leq$ 3.2 psig
d. Steam Line Pressure - Low	4.6	1.31	1.5	$\geq$ 725 psig	$\geq$ 694 psig*
e. Steam Line Pressure-Negative Rate - High	8.0	0.5	0	$\leq$ 100 psi	$\leq$ ~~111.6~~ 122.8 psi**
5. Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level-High-High (P-14)	5.4	2.18	1.5	$\leq$ 82.4% of narrow range instrument span	$\leq$ 84.2% of narrow range instrument span
c. T_{avg}-Low	4.0	1.12	1.2	$\geq$ 564°F	$\geq$ 562°F
d. Doghouse Water Level-High	1.0	0	0.5	11 inches above 577' floor level	12 inches above 577' floor level
e. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Auxiliary Feedwater (Continued)					
c. Steam Generator Water Level - Low-Low	15	12.18	1.5	≥ 17% of span from 0% to 30% RTP increasing linearly to ≥54.9 ~~59.9%~~ of span from 30% to 100% RTP	≥ 10.25% of span from 0% to 30% RTP increasing linearly to ≥ 53.2 ~~58.2%~~ of span from 30% to 100% RTP
d. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
e. Loss-of-Offsite Power	N.A.	N.A.	N.A.	≥ 3500 V	≥ 3200 V
f. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low					
1) 1 CAPS 5220, 5221, 5222	N.A.	N.A.	N.A.	≥ 9.6 psig	≥ 9.5 psig
2) 1 CAPS 5230, 5231, 5232	N.A.	N.A.	N.A.	≥ 10 psig	≥ 9.9 psig
9. Containment Sump Recirculation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Refueling Water Storage Tank Level-Low-Low	N.A.	N.A.	N.A.	≥ 120 inches	≥ 114 inches
Coincident With Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

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TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Atmosphere - High Gaseous Radioactivity (Low Range - EMF-39)	S	R	M	All
b. Reactor Coolant System Leakage Detection (Low Range - EMF-38 and Low Range - EMF-39)	S	R	M	1, 2, 3, 4
2. Fuel Storage Pool Areas				
a. High Gaseous Radioactivity (Low Range - EMF-42)	S	R	M	**
b. Criticality-Radiation Level (Fuel Bridge - Low Range - EMF-15)	S	R	M	*
3. Control Room				
Air Intake Radiation Level - High Gaseous Radioactivity - (Low Range - EMF-43 A & B)	S	R	M	All
4. Auxiliary Building Ventilation				
High Gaseous Radioactivity (Low Range - EMF-41 ~~& 36~~)	S	R	M	All
5. Component Cooling Water System (EMF-46 A&B)	S	R	M	All

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TABLE NOTATIONS

* With fuel in the fuel storage pool area.

** With irradiated fuel in the fuel storage pool areas.

TABLE 3.3-8 DRAFT

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. Wind Speed		
a. Meteorological Tower	Nominal Elev. 661'10"	1
b. Meteorological Tower	Nominal Elev. ~~760'~~ 768'10"	1
2. Wind Direction		
a. Meteorological Tower	Nominal Elev. 661'10"	1
b. Meteorological Tower	Nominal Elev. ~~760'~~ 768'10"	1
3. Air Temperature - ΔT		
Meteorological Tower	Nominal Elev. 760'-661'10"	1

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TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Nominal Elev. 661' 10"	D	SA
b. Nominal Elev. ~~760'~~ 768'10"	D	SA
2. Wind Direction		
a. Nominal Elev. 661' 10"	D	SA
b. Nominal Elev. ~~760'~~ 768'10"	D	SA
3. Air Temperature - ΔT		
Nominal Elev. 760' - 661'10"	D	SA

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* At all times except when the isolation valve is closed and locked.
** During WASTE GAS HOLDUP SYSTEM operation.
*** At all times.

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

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

b. Circuit failure (Alarm only), or

c. Instrument indicates a downscale failure (Alarm only).

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annuciation occurs if any of the following conditions exists:

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

b. Circuit failure, or

c. Instrument indicates a downscale failure.

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

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples in accordance with the manufacturer's recommendations. In addition, a standard gas sample of nominal four volume percent hydrogen, balance nitrogen, shall be used in the calibration to check linearity of the hydrogen analyzer.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples in accordance with the manufacturer's recommendations. In addition, a standard gas sample of nominal four percent oxygen, balance nitrogen, shall be used in the calibration to check linearity of the oxygen analyzer.

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3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

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

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

e ~~d~~. The provisions of Specification 3.0.4 are not applicable.

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SURVEILLANCE REQUIREMENTS (Continued)

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, 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 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

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

   1) Reactor-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A main steam line or feedwater line break.

TABLE 3.4-2

REACTOR COOLANT SYSTEM

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CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	≤ 0.10 ppm	≤ 1.00 ppm
Chloride	≤ 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

*Limit not applicable with T_{avg} less than or equal to 250°F.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

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3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The discharge isolation valve open,

b. A contained borated water volume of between 7743 and 7965 gallons,

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between ~~400~~ 406 and ~~454~~ 460 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each cold leg injection accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

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LIMITING CONDITION FOR OPERATION

3.5.1.2 Each Upper Head Injection Accumulator System shall be OPERABLE with:

a. The discharge isolation valves open,

b. A minimum contained borated water volume of 1807 cubic feet,

c. A boron concentration of between 1900 and 2100 ppm, and

d. The nitrogen-bearing accumulator pressurized to between ~~1206~~ 1210 and ~~1264~~ 1270 psig.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With the Upper Head Injection Accumulator System inoperable, except as a result of closed isolation valve(s), restore the Upper Head Injection Accumulator System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the Upper Head Injection Accumulator System inoperable due to the isolation valve(s) being closed, either immediately open the isolation valve(s) or be in HOT STANDBY within 6 hours and be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.2 Each Upper Head Injection Accumulator System shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water level in the surge tank and nitrogen pressure in the accumulators, and

2) Verifying that each accumulator discharge isolation valve is open.

*Pressurizer pressure above 1900 psig.

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SURVEILLANCE REQUIREMENTS (Continued)

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 333 gpm, and

b) The total pump flow rate is less than or equal to ~~550~~ 565 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 462 gpm, and

b) The total pump flow rate is less than or equal to 660 gpm.

3) For residual heat removal pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 3648 gpm.

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LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1) Less than or equal to L_a, 0.20% by weight of the containment air per 24 hours at P_a, 14.7 psig, or

2) Less than or equal to L_t, ~~118%~~ .118% by weight of the containment air per 24 hours at a ~~reduced~~ pressure of P_t, 7.3 psig.

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

c. A combined bypass leakage rate of less than 0.07 L_a for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths when pressurized to P_a.

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

ACTION:

With: (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) the combined bypass leakage rate exceeding 0.07 L_a, restore the overall integrated leakage rate to less than 0.75 L_a or less than 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than 0.60 L_a, and the combined bypass leakage rate to less than 0.07 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (s)
3. Manual		
NC-141	NC Pump H_2 Drain Tank Pump Discharge	N.A.
NC-142	NC Pump H_2 Drain Tank Pump Discharge	N.A.
NI-3	Boron Injection Tank Line to Cold Legs	N.A.
FW-11	Refueling Water Pump Suction	N.A.
FW-13	Refueling Water Pump Suction	N.A.
CF-91#	Feedwater 1A	N.A.
CF-93#	Feedwater 1B	N.A.
CF-95#	Feedwater 1C	N.A.
CF-97#	Feedwater 1D	N.A.
CA-121#	Aux. Feedwater 1A	N.A.
BW-1#	Aux. Feedwater 1A	N.A.
CA-120#	Aux. Feedwater 1B	N.A.
BW-26#	Aux. Feedwater 1B	N.A.
CA-119#	Aux. Feedwater 1C	N.A.
BW-17#	Aux. Feedwater 1C	N.A.
CA-118#	Aux. Feedwater 1D	N.A.
BW-10#	Aux. Feedwater 1D	N.A.
SM-16#	Main Steam 1A	N.A.
SM-73#*	Main Steam 1A	N.A.
SM-105#	Main Steam 1A	N.A.
SM-121#	Main Steam 1A	N.A.
SM-143#	Main Steam 1A	N.A.
SM-72#*	Main Steam 1B	N.A.
SM-104#	Main Steam 1B	N.A.
SM-120#	Main Steam 1B	N.A.
SM-142#	Main Steam 1B	N.A.
SM SM-1#	Main Steam 1B	N.A.
SM-17#	Main Steam 1B	N.A.
SM-18#	Main Steam 1C	N.A.
SM-71#*	Main Steam 1C	N.A.

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PLANT SYSTEMS

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AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY witin 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

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

1) Verifying that each motor-driven pump develops a total dynamic head of greater than or equal to 3210 feet at a flow of greater than or equal to 500 gpm;

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2) Verifying that the steam turbine-driven pump develops a total dynamic head of greater than or equal to 3217 feet at a flow of greater than or equal to 100 gpm when the secondary steam supply pressure is greater than 600 psig and the auxiliary feedwater pump turbine is operating at 3600 rpm. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

PLANT SYSTEMS

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3/4.7.13 GROUNDWATER LEVEL

LIMITING CONDITION FOR OPERATION

3.7.13 The groundwater level shall be maintained at or below the top of the adjacent floor slabs of the Reactor Containment Building and the Auxiliary Building.

APPLICABILITY: At all times.

ACTION:

a. With the groundwater level above the top of the adjacent floor slab by less than or equal to 5 feet, reduce the groundwater level to or below the top of the affected adjacent floor slab within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the groundwater level above the top of the adjacent floor slab by greater than 5 feet but less than 15 feet, reduce the groundwater level to less than or equal to 5 feet above the top of the affected adjacent floor slab within 24 hours and to or below the top of the affected adjacent floor slab within 7 days of initially exceeding the above limits or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the groundwater level above the top of the adjacent floor slab by greater than or equal to 15 feet, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours. Perform an engineering evaluation ~~of~~ to determine the effects of this higher groundwater level on the affected building(s) and submit the results of this evaluation and any corrective action determined necessary to the Commission as a Special Report pursuant to Specification 6.9.2 prior to increasing T_{avg} above 200°F.

d. Determine the rate of rise of groundwater when the level reaches the top of the floor slab. If the rate of rise of the groundwater level is greater than or equal to 0.3 feet per hour, determine the rate of rise at least once per 30 minutes. If the rate of rise exceeds 0.5 feet per hour for more than 1 hour, be in at least HOT STANDBY within 1 hour and in COLD SHUTDOWN within the following 30 hours. If the rate of rise is less than 0.5 feet per hour, comply with the requirements of ACTIONS a. b. and c. above.

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SURVEILLANCE REQUIREMENTS (Continued)

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected blackout loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the blackout loads. After energization, the

steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an ESF Actuation test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be at 4160 ± 420 volts and 60 ± 1.2 Hz within 11 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, and

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

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test; and

   c) Verifying that all automatic diesel generator trips, except engine overspeed, low-low lube oil pressure, generator differential, and the 2 out of 3 voltage controlled overcurrent relay scheme, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.

7) Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 7700 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 7000 kW. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 11 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
1. 6900 VAC Swgr	
Primary Bkr RCP1A Backup Bkr 1TA-3	Reactor Coolant Pump 1A
Primary Bkr RCP1B Backup Bkr 1TB-3	Reactor Coolant Pump 1B
Primary BKR RCP1C Backup Bkr 1TC-3	Reactor Coolant Pump 1C
Primary BKR RCP1D Backup Bkr 1TD-3	Reactor Coolant Pump 1D
2. 600 VAC MCC	
1EMXC-F01B Primary Bkr Backup Fuse	Accumulator 1B Discharge Isol Vlv 1NI76A
1EMXC-F01C Primary Bkr Backup Fuse	Check Valve Test Header Cont Isol Vlv 1NI95A
1EMXC-F02A Primary Bkr Backup Fuse	Train A Alternate Power To ND LTDN Vlv 1ND1B
1EMXC-F02B Primary Bkr Backup Fuse	Hot Leg Inj. Check Vlv Test Isol Vlv
1EMXC-F02C Primary Bkr Backup Fuse	Cont Isol at 134 Deg. Annulus Area Vlv 1VX312A I
1EMXC-F03A Primary Bkr Backup Fuse	NC Pump 1C Thermal Barrier Outlet Isol Vlv 1KC345A
1EMXC-F03B Primary Bkr Backup Fuse	N_2 to Prt Cont Isol Inside Vlv 1NC54A
1EMXC-F03C Primary Bkr Backup Fuse	Pressurizer Power-Operated Relief Isol Vlv 1NC33A

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TABLE 3.8-1 (Continued) DRAFT

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
2. 600 VAC MCC (Continued)	
1MXM-F08D Primary Bkr Backup Fuse	Ice Condenser Refrigeration Floor Cool Defrost Heater 1A
1MXM-F09A Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 1A1 Fan Motor A & B
1MXM-F09B Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 1B2 Fan Motor A & B
1MXM-F09C Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 1A4 Fan Motor A & B 3
1MXM-F09D Primary Bkr Backup Fuse	Ice Condenser Air Handling Unit 1B4 Fan Motor A & B
1MXM-F10A Primary Bkr Backup Fuse	Containment Floor and Equipment Sump Pump Motor 1A1
1MXM-F10B Primary Bkr Backup Fuse	Containment Floor and Equipment Sump Pump Motor 1B1
1MXN-F01F Primary Bkr Backup Fuse	Stud Tensioner Hoist 1B
1MXN-F02A Primary Bkr Backup Fuse	NC Pump 1B Oil Lift Pump Motor 2
1MXN-F02B Primary Bkr Backup Fuse	NC Pump 1C Oil Lift Pump Motor 2
1MXN-F02E Primary Bkr Backup Fuse	Stud Tensioner Hoist 1C

TABLE 3.8-1 (Continued) DRAFT

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
3. 600 VAC Pressurizer Heater Power Panels (Continued)	
PHP1A-F01C Primary Bkr Backup Fuse	Pressurizer Heaters 9, 10 & 32
PHP1A-F02C Primary Bkr Backup Fuse	Pressurizer Heaters 11, 12 & 35
PHP1A-F02D Primary Bkr Backup Fuse	Pressurizer Heaters 13, 14 & 37
PHP1A-F02E Primary Bkr Backup Fuse	Pressurizer Heaters 17, 18 & 42
PHP1~~A~~B-F01A Primary Bkr Backup Fuse	Pressurizer Heaters 21, 47 & 48
PHP1B-F01B Primary Bkr Backup Fuse	[illegible]r Heaters [illegible] & 54
PHP1B-F01C Primary Bkr Backup Fuse	Pressurizer Heaters 31, 59 & 60
PHP1B-F02C Primary Bkr Backup Fuse	Pressurizer Heaters 36, 65 & 66
PHP1B-F02D Primary Bkr Backup Fuse	Pressurizer Heaters 41, 71 & 72
PHP1B-F02E Primary Bkr Backup Fuse	Pressurizer Heaters 46, 77 & 78
PHP1C-F01A Primary Bkr Backup Fuse	Pressurizer Heaters 7, 8 & 30

TABLE 3.8-1 (Continued) DRAFT

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	SYSTEM POWERED
3. 600 VAC Pressurizer Heater Power Panels (Continued)	
PHP1~~A~~D-F02E Primary Bkr Backup Fuse	Pressurizer Heaters 43, 73 & 74
4. 250 VDC Reactor Building Deadlight Panelboard	
1DLD-2 Primary Bkr Backup Fuse	Lighting Panelboard No. 1LR1, 1LR2, 1LR3, 1LR4
1DLD-3 Primary Bkr Backup Fuse	Lighting Panelboard No. 1LR13, 1LR14
1DLD-4 Primary Bkr Backup Fuse	Lighting Panelboard No. 1LR5, 1LR6
1DLD-5 Primary Bkr Backup Fuse	Lighting Panelboard No. 1LR10, 1LR11
1DLD-10 Primary Bkr Backup Fuse	Lighting Panelboard No. 1LR8
5. 120 VAC Panelboards	
1ELB-5 Primary Bkr Backup Fuse	Emergency A.C. Lighting
1ELB-7 Primary Bkr Backup Fuse	Emergency A.C. Lighting
1ELB-13 Primary Bkr Backup Fuse	Emergency A.C. Lighting
1ELB-15 Primary Bkr Backup Fuse	Emergency A.C. Lighting

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SURVEILLANCE REQUIREMENTS (Continued)

4.9.4.2 The Reactor Building Containment Purge System shall be demonstrated OPERABLE:

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

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

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedures guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 28,000 cfm ± 10% (both exhaust fans operating);

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

   3) Verifying a system flow rate of 28,000 cfm ± 10% (both exhaust fans operating) during system operation when tested in accordance with ANSI N510-1980.

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

d. At least once per 18 months by:

   1) Verifying that the pressure drop across the combined HEPA filters, charcoal adsorber banks, and moisture separators is less than 8 inches Water Gauge while operating the system at a flow rate of 28,000 cfm ± 10% (both exhaust fans operating);

   2) Verifying that the filter cooling bypass valves can be opened by operator action; and

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3/.11.3 SOLID RADIOACTIVE WASTES

LIMIITNG CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be solidified or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures and/or the Solid Radwaste System as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM:

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

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.12, to assure SOLIDIFICATION of subsequent batches of waste; and 13

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

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3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations shall be made including direct radiation contributions from the units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

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

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LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant
14 to Specification 6.13, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

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TABLE 4.12-1 (Continued)

TABLE NOTATIONS

(1) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.6.

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3) The LLD is defined, for purposes of these specifications, as the smallest concentrations of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

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

Where:

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

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

E = the counting efficiency (counts per disintegration),

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

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical yield, when applicable,

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

Δt = the elapsed time between environmental collection, or end of the sample collection period, and time of counting (s).

Typical values of E, V, Y and Δt should be used in the calculation.

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AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure *3.2-1* while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least two of four or two of three OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and REACTOR COOLANT SYSTEM FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, coolant flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

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

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

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

Reactor Coolant system

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

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed. The rod bow penalty as a function of burnup applied for $F^N_{\Delta H}$ is calculated with the methods described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979, and the maximum rod bow penalty is 2.7% DNBR. Since the safety analysis is performed with plant-specific safety DNBR limits of 1.49 and 1.47 compared to the design DNBR limits of 1.34 and 1.32, respectively, for the typical and thimble cells, there is a 10% thermal margin available to offset the rod bow penalty of 2.7% DNBR

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

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

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response time.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) nuclear service water pumps start and automatic valves position, and (11) component cooling pumps start and automatic valves position.

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3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain calculated DNBR above the design DNBR value during Conditions I and II events. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 1 hour. 6 hours

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or residual heat removal loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either residual heat removal or Reactor Coolant System) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single residual heat removal loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two residual heat removal loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one residual heat removal pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump below P-7 with one or more cold legs less than or equal to ~~300°F~~ 285°F are provided to prevent pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The Reactor Coolant System will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the Reactor Coolant System from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating residual heat removal loop, connected to the Reactor Coolant System, provides overpressure

PRESSURE/TEMPERATURE LIMITS (Continued)

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

COOLDOWN

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

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T

PRESSURE/TEMPERATURE LIMITS (Continued)

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile 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 PORVs or a Reactor Coolant System vent opening of at least 4.5 square inches ensures that the Reactor Coolant System 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 cold legs are less than or equal to 285 ~~300~~°F. Either PORV has adequate relieving capability to protect the Reactor Coolant System from overpressurization when the transient is limited to either: (1) the start of an idle reactor coolant pump with the secondary water temperature of the steam generator less than or equal to 50°F above the cold leg temperatures, or (2) the start of a Safety Injection pump and its injection into a water solid Reactor Coolant System.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

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ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump and one Safety Injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE centrifugal charging pump to be inoperable below ~~[illegible]~~ 285 °F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK

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

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the refueling water storage tank also ensure a pH value of between 8.5 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

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

3/4.6.1.2 CONTAINMENT LEAKAGE

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

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

3/4.6.1.3 CONTAINMENT AIR LOCKS

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

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 1.5 psig, and (2) the containment peak pressure does not exceed the design pressure of 15 psig during LOCA conditions.

3/4.6.1.8 ANNULUS VENTILATION SYSTEM

The OPERABILITY of the Annulus Ventilation System ensures that during LOCA conditions, containment vessel leakage into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere. Operation of the system with the heaters operating to maintain low humidity using automatic control for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. This requirement is necessary to meet the assumptions used in the safety analyses and limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during LOCA conditions. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

3/4.6.1.9 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust isolation valves for the lower compartment (24-inch), and instrument room (12-inch), and the Hydrogen Purge System (4-inch) are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA. Maintaining these valves sealed closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 24-inch purge supply and exhaust isolation valves for the upper compartment and the 4-inch Containment Air Release and Addition System valves since, unlike the lower compartment, instrument room, and the Hydrogen Purge System valves, these 24-inch valves and 4-inch valves are capable of closing during a LOCA. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment purging operation. Operation with the line open will be limited to 250 hours during a calendar year for the 24-inch valves and 2000 hours during a calendar year for the 4-inch valves. The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to support the additional time requests.

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

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FIRE SUPPRESSION SYSTEMS (Continued)

In the event the Fire Suppression Water System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.11 FIRE BARRIER PENETRATIONS

The functional integrity of the fire barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. These design features minimize the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishing of the fire. The fire barrier penetrations are a passive element in the facility fire protection program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors, fire dampers, and other fire barriers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, either: (1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or (2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established, until the barrier is restored to functional status.

3/4.7.12 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ± 3.9°F.

3/4.7.13 GROUNDWATER LEVEL

This specification is provided to ensure tha~~x~~ t groundwater levels will be monitored and prevented from rising to unacceptable levels. High groundwater levels could result in unacceptable structural stresses in the Containment and/or Auxiliary Building due to uplift and hydrostatic forces during design basis events. Although these buildings have been statically analyzed to withstand soil pressure along with the uplift and hydrostatic forces resulting from groundwater rebound to yard elevation (593'6"), this analysis did not include any other loadings and was not a design condition for these buildings.

BASES

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

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

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term, verify, as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

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

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979. If any other metallic structures (building, new or modified piping systems, conduits) are placed in the ground near the Fuel Oil Storage System or if the original system is modified, the adequacy and frequency of inspections for the Cathodic Protection System shall be reevaluated and adjusted in accordance with the manufacturer's recommendations.

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

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

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

Table 4.8-3 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.1~~2~~3 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.1~~2~~3 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

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

DOSE (Continued)

Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

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BASES

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM (Continued)

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. ~~For determining conformance to LCOs, these allocations from shard Radwaste Treatment System~~. For determining conformance to LCOs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the WASTE GAS HOLDUP SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. When shared Radwaste Treatment Systems are used by more than one unit on a site, the wastes from all units are mixed for shared treatment; by such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flow rates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste producing units sharing the Radwaste Treatment System. ~~For determining conformance to LCOs, these allocations from shard Radwaste Treatment System.~~ For determining conformance to LCCs, these allocations from shared Radwaste Treatment Systems are to be added to the releases specifically attributed to each unit to obtain the total releases per unit.

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5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-2.

MAPS DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figures 5.1-3 and 5.1-4.

The definition of UNRESTRICTED AREA used in implementing these Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment structure is comprised of a steel containment vessel surrounded by a concrete containment having the following design features:

a. Containment Vessel

1) Nominal inside diameter = 115 feet.
2) Nominal inside height = 171 feet.
3) Nominal thickness of vessel walls = 0.75 inch.
4) Nominal thickness of vessel dome = 0.6875 inch.
5) Nominal thickness of vessel bottom = 0.25 inch.
6) Net free volume = 1.2×10^6 cubic feet.

b. Reactor Building

1) Nominal Annular space = 6 feet.
2) Annulus nominal volume = 484,090 cubic feet.
3) Nominal outside height (top of foundation base to top of dome) = 177 feet.
4) Nominal inside diameter = 127 feet.
5) Minimum cylinder wall thickness = 3 feet.
6) Minimum dome thickness = 2.25 feet.
7) Dome inside radius = 87 feet.

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DESIGN FEATURES

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment vessel is designed and shall be maintained for a maximum internal pressure of 15 psig and a temperature of 328°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1619 grams uranium. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorb~~ed~~er material of which 102 inches shall be 100% boron carbide and remaining 40-inch tip shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure a 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 12,040 ± 100 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown in Figure 5.1-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

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

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SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form on an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation". Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PCP and to the ODCM, pursuant to Specifications 6.~~12~~ 13 and 6.~~13~~ 14, respectively, as well as any major changes to Liquid, Gaseous or Solid Radwaste Treatment Systems, pursuant to Specification 6.~~14~~ 15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified

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*In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

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HIGH RADIATION AREA (Continued)

6.12.2 In addition to the requirements of Specification 6.~~10~~ 12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control ~~or~~ of the Shift Supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

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

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:

a. 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:

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

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

3) Documentation of the fact that the change has been reviewed and found acceptable by the Station Manager.

b. Shall become effective upon review and acceptance by a qualified individual/organization.