



UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
License No. NPF-42

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment to the Wolf Creek Generating Station (the facility) Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated May 24, 1994, as supplemented by letter dated April 6, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

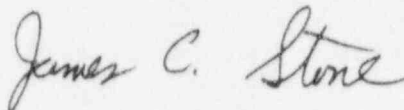
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-42 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 89, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



James C. Stone, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical
Specifications

Date of Issuance: October 2, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

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1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

DEFINITIONS

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

- a. All penetrations required to be closed during accident conditions are either:
 - 1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or
 - 2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.
- b. All equipment hatches are closed and sealed,
- c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,
- d. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE, and
- e. The containment leakage rates determined by Specification 4.b.1.1.d are within the limits listed in the Bases of Specification 3/4.6.1.1.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow from the reactor coolant pump seals.

CORE ALTERATION

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

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these operating limits is addressed in individual Specifications.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.3% $\Delta k/k$.

APPLICABILITY: MODES 3, 4 and 5.

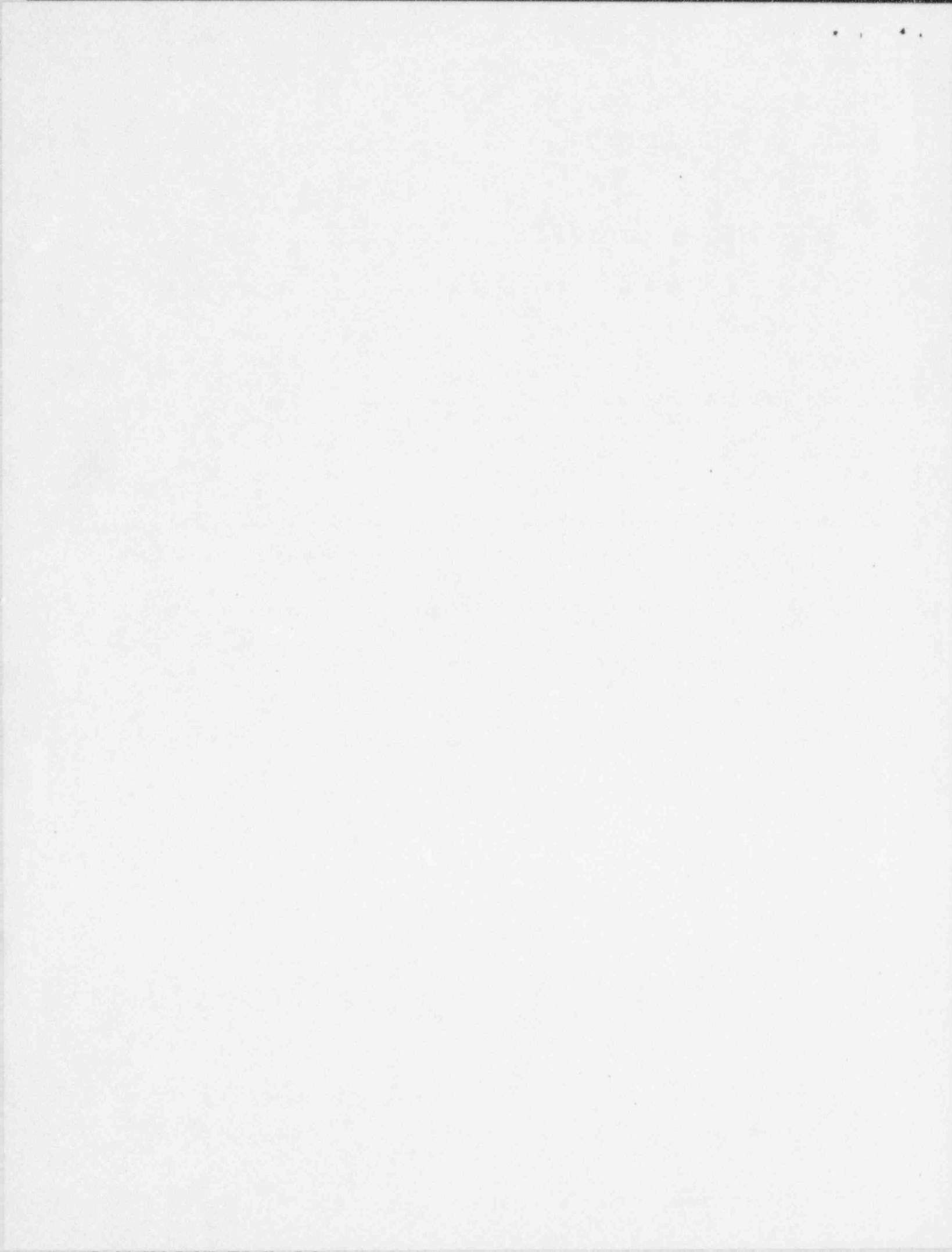
ACTION:

With the SHUTDOWN MARGIN less than 1.3% $\Delta k/k$, within 15 minutes 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% $\Delta k/k$:

- a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);
- b. At least once per 24 hours by consideration of the following factors:
 - 1) Reactor Coolant System boron concentration,
 - 2) Control rod position,
 - 3) Reactor Coolant System average temperature,
 - 4) Fuel burnup based on gross thermal energy generation,
 - 5) Xenon concentration, and
 - 6) Samarium concentration.



REACTIVITY CONTROL SYSTEMS

CORE REACTIVITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The measured core reactivity shall be within $\pm 1\% \Delta k/k$ of predicted values.

APPLICABILITY: MODES 1 and 2

ACTION:

With the measured core reactivity not within limits, within 72 hours:

- a. reevaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation, and
- b. establish appropriate administrative operating restrictions and surveillance requirements, or
- c. be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.5.1 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm 1\% \Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.b. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

4.1.1.5.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to $1.3\% \Delta k/k$ prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.b, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION: The ACTION to be taken is based on the cause of inoperability of control rods as follows:

CAUSE OF INOPERABILITY	ACTION	
	One Rod	More Than One Rod
a) Immovable as a result of excessive friction or mechanical interference or known to be untrippable.	(1)	(1)
b) Misaligned from its group step counter demand height or from any other rod in its group by more than ± 12 steps (indicated position).	(3)	(2)
c) Inoperable due to a rod control urgent failure alarm or other electrical problem in the rod control system, but trippable.	(4)	(4)

- ACTION 1 -
1. Determine that the SHUTDOWN MARGIN is greater than or equal to 1.3% $\Delta k/k$, with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s), is satisfied within 1 hour, and
 2. Be in HOT STANDBY within 6 hours.

ACTION 2 - Be in HOT STANDBY within 6 hours.

ACTION 3 - 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 Specification 3.1.3.6. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

3. The rod is declared inoperable and the SHUTDOWN MARGIN is greater than or equal to 1.3% $\Delta k/k$. 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) A power distribution map is obtained from the movable incore detectors and $F_o(Z)$ and F_{AH}^N are verified to be within their limits within 72 hours; and
 - c) 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.

ACTION 4 - Restore the inoperable rods to OPERABLE status within 72 hours or be in HOT STANDBY within the next 6 hours.

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.

4.1.3.1.3 Prior to reactor criticality, the rod drop time of the individual full-length shutdown and control rods from the fully withdrawn position shall be demonstrated to be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry, with $T_{avg} \geq 551^\circ F$, and all reactor coolant pumps operating:

- a. For all rods following each removal of the reactor vessel head, and
- b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE FULL-
LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

- a. With a maximum of one digital rod position indicator per bank inoperable either:
 1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or
 2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.
- b. With more than one digital rod position indicator per bank inoperable either:
 - 1.a) Determine the position of the nonindicating rods indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, and
 - b) Place the control rods under manual control, and
 - c) Monitor and record Reactor Coolant System average temperature (T_{avg}) at least once per hour, and
 - d) Restore the digital rod position indicators to OPERABLE status within 24 hours such that a maximum of one digital rod position indicator per bank is inoperable, or
2. Be in HOT STANDBY within the next 6 hours.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION (continued)

- c. With a maximum of one demand position indicator per bank inoperable either:
 1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or
 2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

- a. Restore the rod to within the insertion limit specified in the COLR,
or
- b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR:

- a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and
- b. At least once per 12 hours thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

- a. Within 1 hour, verify that the SHUTDOWN MARGIN is greater than or equal to 1.3% $\Delta k/k$ or initiate boration until the SHUTDOWN MARGIN is restored to greater than or equal to 1.3% $\Delta k/k$, and
- b. Restore the control banks to within the limits within 2 hours, or
- c. 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 insertion limits specified in the COLR, or
- d. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6.1 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.

4.1.3.6.2 When in Mode 2 with K_{eff} less than 1, verify that the predicted critical control rod position is within insertion limits within 4 hours prior to achieving reactor criticality.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown monitoring instrumentation channels given in Table 3.3-9 and the auxiliary shutdown panel (ASP) controls shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With the ASP controls inoperable, restore the inoperable ASP controls to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies given in Table 4.3-6.

4.3.3.5.2 The ASP controls shall be demonstrated OPERABLE at least once per 18 months by operating each actuated component from the ASP.

4.3.3.5.3 The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 for the turbine-driven auxiliary feedwater pump or the atmospheric dump valves.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>READOUT LOCATION</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>MINIMUM CHANNELS OPERABLE</u>
1. RCS Pressure-Wide Range	ASP*	2	1
2. Reactor Coolant Temperature-Cold Leg	ASP*	4	1
3. Source Range, Neutron Flux#	ASP*	2	1
4. Reactor Trip Breaker Indication	RTS**	1/trip breaker	1/trip breaker
5. Reactor Coolant Temperature - Hot Leg	ASP*	2	1
6. Reactor Coolant Pump Breakers	***	1/pump	1/pump
7. Pressurizer Pressure	ASP*	1	1
8. Pressurizer Level	ASP*	2	1
9. Steam Generator Pressure	ASP*	2/stm. gen.	1/stm. gen.
10. Steam Generator Level	ASP*	2/stm. gen.	1/stm. gen.
11. Auxiliary Feedwater Flow Rate	ASP*	4	1
12. Auxiliary Feedwater Suction Pressure	ASP*	3	1

*Auxiliary Shutdown Panel.

**Reactor Trip Switchgear.

***13.8 kV Switchgear.

#Not required OPERABLE in MODE 1 or in MODE 2 above P-6 Setpoint.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATIONSURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>
1. RCS Pressure - Wide Range	M	R
2. Reactor Coolant Temperature - Cold Leg	M	R
3. Source Range, Neutron Flux#	M	R
4. Reactor Trip Breaker Indication	M	N.A.
5. Reactor Coolant Temperature - Hot Leg	M	R
6. Reactor Coolant Pump Breakers	N.A.	N.A.
7. Pressurizer Pressure	M	R
8. Pressurizer Level	M	R
9. Steam Generator Pressure	M	R
10. Steam Generator Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Auxiliary Feedwater Pump Suction Pressure	M	R

#Not required OPERABLE in MODE 1 or in MODE 2 above P-6 Setpoint.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 30 days or prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.9.2 within the following 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.
- b. With the number of OPERABLE accident monitoring instrumentation channels, except for instrument functions 10, 16 and 18 (Containment Hydrogen Concentration Level, Containment Radiation Level, and the Reactor Vessel Level Indicating System), less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore one channel to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- c. With the number of OPERABLE channels for instrument functions 16 and 18 (Containment Radiation Level and the Reactor Vessel Level Indicating System), less than the Minimum Channels OPERABLE requirements of Table 3.3-10, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours and either restore one inoperable channel to OPERABLE status within 7 days, or prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.9.2 within the following 14 days outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.
- d. With the number of OPERABLE channels for the containment hydrogen concentration level monitor less than the Minimum Channels OPERABLE requirement of Table 3.3-10, restore one channel to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

<u>INSTRUMENT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>MINIMUM CHANNELS OPERABLE</u>
1. Containment Pressure - Normal Range	2	1
2. Reactor Coolant Outlet Temperature - T _{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T _{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	2/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Containment Hydrogen Concentration Level	2	1
11. Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
12. Deleted		
13. Deleted		
14. Neutron Flux	2	1
15. Containment Water Level	2	1
16. Containment Radiation Level (High Range)	2	1
17. Thermocouple/Core Cooling Detection System	4/core quadrant	2/core quadrant
18. Reactor Vessel Level Indicating System	2	1

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>INSTRUMENT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>
1. Containment Pressure - Normal Range	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Containment Hydrogen Concentration Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Deleted		
13. Deleted		
14. Neutron Flux	M	R ⁽¹⁾
15. Containment Water Level	M	R
16. Containment Radiation Level (High Range)	M	R ⁽²⁾
17. Thermocouple/Core Cooling Detection System	M	R
18. Reactor Vessel Level Indicating System	M	R

⁽¹⁾Neutron detectors may be excluded from channel calibration.

⁽²⁾CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

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REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig \pm 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

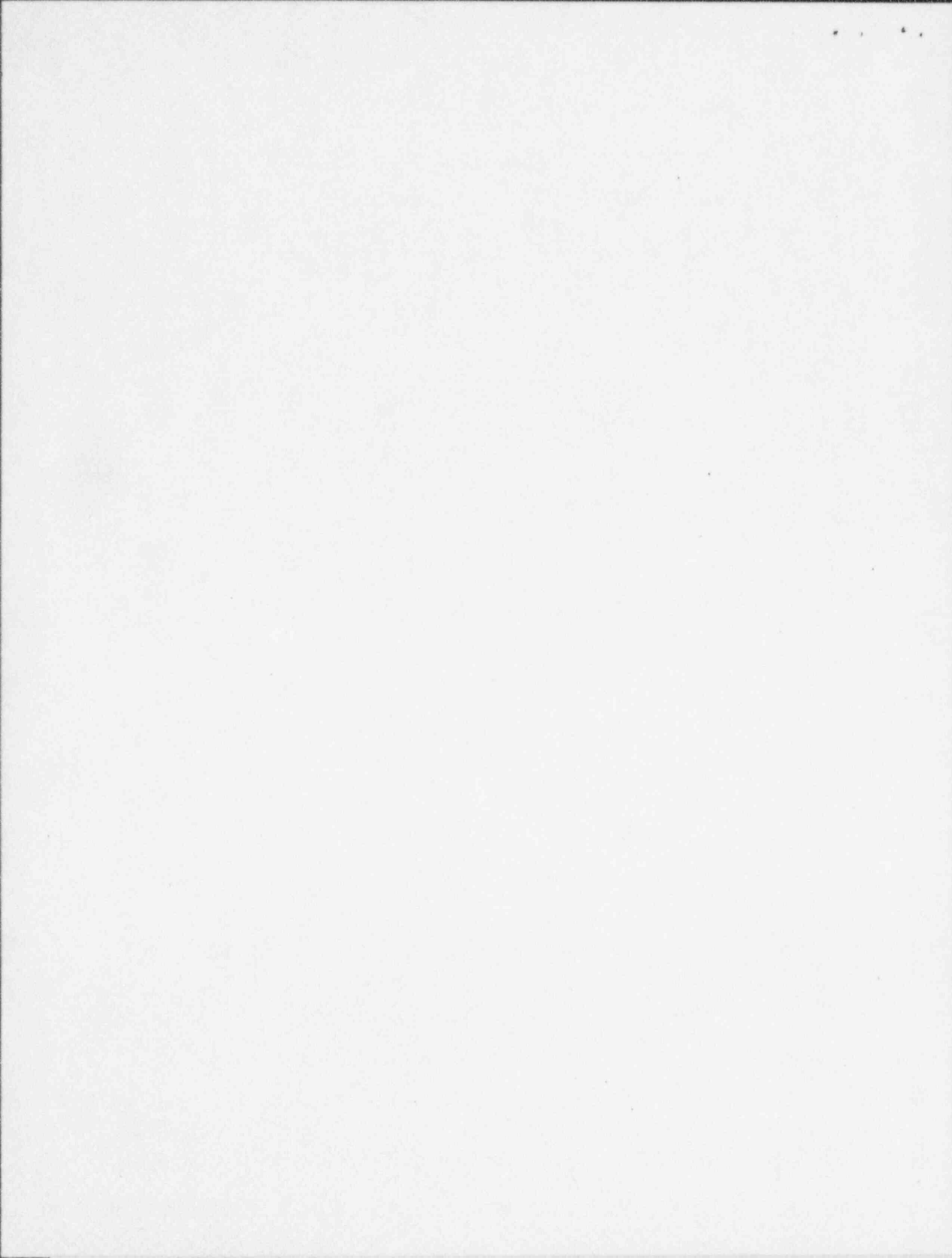
4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

<u>VALVE NUMBER</u>	<u>FUNCTION</u>
BBV8948 A, B, C, D	SI/RHR/Accum. Cold Leg Injection
BBV8949 A, B, C, D	SI/RHR Hot Leg Injection
BBV001, 022, 040, 059	Bit Cold Leg Injection
BBPV8702 A, B	RHR Normal Suction
EJV8841 A, B	RHR Hot Leg Recirc Ctmt ISO
EJHV8701 A, B	RHR Normal Suction
EMV001, 002, 003, 004	SI Hot Leg Inj Ctmt ISO
EM8815	BIT Inj. Ctmt Isolation
EPV010, 020, 030, 040	SI Cold Leg Inj Ctmt ISO
EPV8818, A, B, C, D	RHR Cold Leg Inj Ctmt ISO
EPV8956 A, B, C, D	Accum Inj Isolation



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REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following groups of two overpressure protection devices shall be OPERABLE when the Reactor Coolant System (RCS) is not depressurized through a 2 square inch or larger vent:

- a. Two residual heat removal (RHR) suction relief valves with Setpoints of 450 psig \pm 3%, or
- b. Two power-operated relief valves (PORV) with Setpoints which do not exceed the limit established in Figure 3.4-4, or
- c. One RHR suction relief valve and one PORV with Setpoints as prescribed above.

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

ACTION:

- a. With one of the two required overpressure protection devices inoperable in MODE 3 or 4, restore two overpressure protection devices to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.
- b. With one of the two required overpressure protection devices inoperable in MODES 5 or 6, restore two overpressure protection devices to OPERABLE status within 24 hours, or complete depressurization and venting of the RCS through at least a 2 square inch vent within the next 8 hours.
- c. With both of the two required overpressure protection devices inoperable, complete depressurization and venting of the RCS through at least a 2 square inch vent within 8 hours.
- d. In the event either the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.
- e. The provisions of Specification 3.0.4 are not applicable.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 ECCS SUBSYSTEMS - $T_{avg} < 200^{\circ}F$

LIMITING CONDITION FOR OPERATION

3.5.4 All Safety Injection pumps and one Centrifugal Charging Pump shall be inoperable.

APPLICABILITY: MODE 5 and MODE 6 with the Reactor Vessel head on.*

ACTION:

- a. With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.
- b. With two Centrifugal Charging Pumps OPERABLE, restore one of the Centrifugal Charging Pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 All Safety Injection pumps shall be demonstrated inoperable** by verifying that the motor circuit breakers are secured in the open position at least once per 31 days.

4.5.4.2 One Centrifugal Charging Pump shall be demonstrated inoperable** by verifying that the motor circuit breakers are secured in the open position at least once per 31 days.

* When the RCS water level is below the top of the reactor vessel flange, both Safety Injection Pumps may be OPERABLE for the purpose of protecting the decay heat removal function.

**An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

- a. A minimum contained borated water volume of 394,000 gallons,
- b. A boron concentration of between 2400 and 2500 ppm of boron,
- c. A minimum solution temperature of 37°F, and
- d. A maximum solution temperature of 100°F.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

- a. At least once per 7 days by:
 - 1) Verifying the contained borated water volume in the tank, and
 - 2) Verifying the boron concentration of the water.
- b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 37°F or greater than 100°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

- a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3;
- b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3;
- c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a , 48 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.1d. for all other Type B and C penetrations, the combined leakage rate is less than $0.60 L_a$;
- d. By performing containment leakage rate testing, except for containment air lock testing, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions; and
- e. By verifying containment structural integrity in accordance with the Containment Tendon Surveillance Program of Specification 6.8.5.c.

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

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

- a. Both doors closed except when the air lock is being used for normal transit entry and exits through the containment, then at least one air lock door shall be closed, and
- b. An overall air lock leakage rate of less than or equal to $0.05 L_a$ at P_a , 48 psig.

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

ACTION:

- a. With one containment air lock door inoperable:
 1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,
 2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days,
 3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and
 4. The provisions of Specification 3.0.4 are not applicable.
- b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

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

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

Location

- a. Containment Cooler Inlet located near NNE wall (E1 2068'-8"),
- b. Containment Cooler Inlet located near West wall (E1 2068'-8"),
- c. Containment Cooler Inlet located near NNW wall (E1 2068'-8"), and
- d. Containment Cooler Inlet located near East wall (E1 2068'-8").

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

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

- a. Each 36-inch containment shutdown purge supply and exhaust isolation valve shall be closed and blank flanged, and
- b. The 18-inch containment mini-purge supply and exhaust isolation valve(s) may be open for up to 2000 hours during a calendar year.

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

ACTION:

- a. With a 36-inch containment purge supply and/or exhaust isolation valve open or not blank flanged, close and/or blank flange that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With the 18-inch containment mini-purge supply and/or exhaust isolation valve(s) open for more than 2000 hours during a calendar year, close the open 18-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.
- c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.7.2 and/or 4.6.1.7.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 36-inch containment shutdown purge supply and exhaust isolation valve(s)* shall be verified blank flanged and closed at least once per 31 days.

4.6.1.7.2 Each 36-inch containment shutdown purge supply and exhaust isolation valve and its associated blank flange shall be leak tested at least once per 24 months and following each reinstallation of the blank flange when pressurized to P_a , 48 psig, and verifying that when the measured leakage rate for these valves and flanges, including stem leakage, is added to the leakage rates determined pursuant to Specification 4.6.1.1d., for all other Type B and C penetrations, the combined leakage rate is less than $0.60 L_a$.

4.6.1.7.3 The cumulative time that all 18-inch containment mini-purge supply and/or exhaust isolation valves have been open during a calendar year shall be determined at least once per 7 days.

4.6.1.7.4 At least once per 3 months each 18-inch containment mini-purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than $0.05 L_a$ when pressurized to P_a .

*Except valves and flanges which are located inside containment. These valves shall be verified to be closed with their blank flanges installed prior to entry into MODE 4 following each COLD SHUTDOWN.

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Two independent groups of containment cooling fans shall be OPERABLE with two fan systems to each group.

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

ACTION:

- a. With one group of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore the inoperable group of cooling fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With two groups of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore at least one group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With one group of the above required containment cooling fans inoperable and one Containment Spray System inoperable, restore the inoperable Containment Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

- a. At least once per 31 days by:
 - 1) Starting each non-operating fan group from the control room, and verifying that each fan group operates for at least 15 minutes.
 - 2) Verifying that each valve (manual, power-operated, or automatic) in the cooling water flow path serving the containment coolers that is not locked, sealed, or otherwise secured in position, is in its correct position.
- b. At least once per 18 months by verifying that on a Safety Injection test signal, the fans start in slow speed or, if operating, shift to slow speed and the cooling water flow rate increases to at least 2000 gpm to each cooler group.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

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

ACTION:

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

- a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or
- b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or
- c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or
- d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

<u>PENETRATIONS</u>	<u>VALVE NUMBER</u>	<u>FUNCTION</u>	<u>TYPE LEAK TEST REQUIRED</u>	<u>MAXIMUM ISOLATION TIME (Seconds)</u>
8. Hand-Operated and Check Valves - (Continued)				
P-66	EN V-017	CTMT Spray Pump B to CTMT Spray Nozzles	A	N.A.
P-45	EP V-046	Accumulator Nitrogen Supply Line	C	N.A.
P-43	HD V-016	Auxiliary Steam to Decon System	C	N.A.
P-43	HD V-017	Auxiliary Steam to Decon System	C	N.A.
P-63	KA V-039	Rx Bldg Service Air Supply	C	N.A.
P-63	KA V-118	Rx Bldg Service Air Supply	C	N.A.
P-98	KB V-001	Breathing Air Supply to RX Bldg	C	N.A.
P-98	KB V-002	Breathing Air Supply to RX Bldg	C	N.A.
P-30	KA V-204	Rx Bldg Instrument Air Supply	C	N.A.
P-67	KC V-478	Fire Protection Supply to RX Bldg	C	N.A.
P-57	SJ V-111	Liquid Sample from PASS to RCDT	A,C	N.A.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

<u>PENETRATIONS</u>	<u>VALVE NUMBER</u>	<u>FUNCTION</u>	<u>TYPE LEAK TEST REQUIRED</u>	<u>MAXIMUM ISOLATION TIME (Seconds)</u>
9. Other Automatic Valves				
P-1	AB-HV-11***	Mn. Stm. Isol.	A	N.A.
P-2	AB-HV-14***	Mn. Stm. Isol.	A	N.A.
P-3	AB-HV-17***	Mn. Stm. Isol.	A	N.A.
P-4	AB-HV-20***	Mn. Stm. Isol.	A	N.A.
P-5	AE-FV-42***	Mn. FW Isol.	A	N.A.
P-6	AE-FV-39***	Mn. FW Isol.	A	N.A.
P-7	AE-FV-40***	Mn. Fw Isol.	A	N.A.
P-8	AE-FV-41***	Mn. FW Isol.	A	N.A.
P-9	BM-HV-4**	SG Blowdn. Isol.	A	10
P-10	BM-HV-1**	SG Blowdn. Isol.	A	10
P-11	BM-HV-2**	SG Blowdn. Isol.	A	10
P-12	BM-HV-3**	SG Blowdn. Isol.	A	10

**The provisions of Specification 3.0.4 are not applicable.

***These valves are included for table completeness. The requirements of Specification 3.6.3 do not apply; instead, the requirements of Specification 3.7.1.5, 3.7.1.7 and Specification 3.3.2 apply to the Main Steam Isolation Valves and Main Feedwater Isolation Valves.

CONTAINMENT SYSTEMS

HYDROGEN CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.4.2 A Hydrogen Control System shall be OPERABLE with two independent Hydrogen Recombiner Systems.

APPLICABILITY: MODES 1 and 2

ACTION:

With one of the two independent Hydrogen Recombiner Systems inoperable, restore the inoperable Hydrogen Recombiner System to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

- a. At least once per 6 months by verifying, during a Hydrogen Recombiner System functional test, that the heater air temperature increases to greater than or equal to 1150°F within 5 hours; and
- b. At least once per 18 months by:
 - 1) Performing a CHANNEL CALIBRATION of all hydrogen recombiner system instrumentation and control circuits,
 - 2) Verifying through a visual examination that there is no evidence of abnormal conditions within the hydrogen recombiner system enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and
 - 3) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

PLANT SYSTEMS

MAIN FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.7 Each main feedwater isolation valve (MFIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

MODES 1 and 2: With one MFIV inoperable but open, operation may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in HOT STANDBY within the next 6 hours.

MODE 3: With one MFIV inoperable, subsequent operation in MODE 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.7 Each MFIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

PLANT SYSTEMS

3/4.7.7 EMERGENCY EXHAUST SYSTEM - AUXILIARY BUILDING

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent Emergency Exhaust Systems shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.7 Each Emergency Exhaust System shall be demonstrated OPERABLE:

- a. By performing surveillance requirements 4.9.13.a through 4.9.13.f, and,
- b. At least once per 18 months by:
 - 1) Verifying that the system maintains the Auxiliary Building at a negative pressure of greater than or equal to $\frac{1}{4}$ inch Water Gauge relative to the outside atmosphere during system operation,
 - 2) Verifying that the system starts on a Safety Injection test signal.

REFUELING OPERATIONS

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

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

- a. The equipment door closed and held in place by a minimum of four bolts,
- b. A minimum of one door in each airlock is closed*, and
- c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:
 - 1) Closed by an isolation valve, blind flange, manual valve, or approved functional equivalent, or
 - 2) Be capable of being closed by an OPERABLE automatic containment purge isolation valve.

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

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

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

- a. Verifying the penetrations are in their closed/isolated condition, or
- b. Testing the containment purge isolation valves per the applicable portions of Specification 4.6.3.2.

*An emergency escape hatch temporary closure device is an acceptable replacement for that airlock door.

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REFUELING OPERATIONS

3/4.9.11 WATER LEVEL-STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

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

ACTION:

- a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.
- b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

SPECIAL TEST EXCEPTIONS

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

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

- a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and
- b. The limits of Specifications 3.2.2, 3.2.3, and 3.2.5.c are maintained and determined at the frequencies specified in Specification 4.10.2.2, below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2, 3.2.3, or 3.2.5.c being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

- a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2, 3.2.3, and 3.2.5.c, or
- b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

- a. Specification 4.2.2.2,
- b. Specification 4.2.3.2, and
- c. Specification 4.2.5.2.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6, may be suspended during the performance of PHYSICS TESTS provided:

- a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,
- b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and
- c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

- a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.
- b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of the following requirements may be suspended:

- a. Specification 3.2.3, 3.2.5.c and 3.4.1.1 - During the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:
 - 1) The THERMAL POWER does not exceed the P-10 Interlock Setpoint, and
 - 2) The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.
- b. Specification 3.4.1.2 - During the performance of hot rod drop time measurements in MODE 3 provided at least three reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During operation below the P-10 Interlock Setpoint or performance of hot rod drop time measurements.

ACTION:

- a. With the THERMAL POWER greater than the P-10 Interlock Setpoint during the performance of startup and PHYSICS TESTS, immediately open the Reactor trip breakers.
- b. With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately place two reactor coolant loops in operation.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-10 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-10 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

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

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3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg} . The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1.3% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

REACTIVITY CONTROL SYSTEMS

BASES

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC End of Life (EOL) value specified in the CORE OPERATING LIMITS REPORT (COLR). The 300 ppm surveillance limit MTC value specified in the COLR represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting EOL MTC value specified in the COLR.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{MDT} temperature.

3/4.1.1.5 CORE REACTIVITY

The core is considered to be operating within acceptable design limits when measured core reactivity is within $\pm 1\% \Delta k/k$ of the predicted value at steady state thermal conditions. Deviations from the design limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations. The difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the design limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples. Therefore, spurious violations of the design limit due to uncertainty in measuring the RCS boron concentration are unlikely.

The acceptance criteria for core reactivity ($\pm 1\% \Delta k/k$ of the predicted value) ensures plant operation is maintained within the assumptions of the safety analyses.

REACTIVITY CONTROL SYSTEMS

BASES

CORE REACTIVITY (Continued)

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Therefore, every accident evaluation is dependent upon accurate evaluation of core reactivity. SDM and reactivity transients, such as control rod withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle. These are used to predetermine reactivity behavior and RCS boron concentration requirements for reactivity control during fuel depletion.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted RCS boron concentrations for identical core conditions at beginning of life (BOL) do not agree, then the assumptions used in the reload cycle design analysis or requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOL, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted boron letdown curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOL, or that an unexpected change in core conditions has occurred.

The normalization of predicted RCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. The normalization is performed at BOL conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the cycle.

Should an anomaly develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required completion time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

REACTIVITY CONTROL SYSTEMS

BASES

CORE REACTIVITY (Continued)

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core conditions at the time of RCS boron concentration sampling, then a recalculation of the RCS boron concentration requirements may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If operational restriction or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required completion time of 72 hours is adequate for preparing whatever operating restrictions or surveillances that may be required to allow continued operation.

3/4/1.2 DELETED

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and 18, 210 and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

For purposes of determining compliance with Specification 3.1.3.1, any immovability of a control rod invokes ACTION Statement 3.1.3.1.a. Before utilizing ACTION Statement 3.1.3.1.c, the rod control urgent failure alarm must be illuminated or an electrical problem must be detected in the rod control system. The rod is considered trippable if the rod was demonstrated OPERABLE during the last performance of Surveillance Requirement 4.1.3.1.2 and met the rod drop time criteria during the last performance of Surveillance Requirement 4.1.3.1.3.

REACTIVITY CONTROL SYSTEMS

BASES

MOVABLE CONTROL ASSEMBLIES (Continued)

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The power reduction and shutdown time limits given in ACTION statements 3.1.3.2.a.2, 3.1.3.2.b.2, and 3.1.3.2.c.2, respectively, are initiated at the time of discovery that the compensatory actions required for POWER OPERATION can no longer be met.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

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

INSTRUMENTATION

BASES

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

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 times.

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 System isolates, (4) the emergency diesel generators start, (5) containment spray pumps start and automatic valves position, (6) containment isolates, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, (11) essential service water pumps start and automatic valves position, and (12) isolate normal control room ventilation and start Emergency Ventilation System.

Engineered Safety Features Actuation System Interlocks

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressure P-11 automatically reinstates safety injection actuation on low pressurizer pressure and low steamline pressure and automatically blocks steamline isolation on negative steamline pressure rate. On decreasing pressure; P-11 allows the manual block of Safety Injection on low pressurizer pressure and low steamline pressure and allows steamline isolation on negative steamline pressure rate to become active upon manual block of low steamline pressure SI.

INSTRUMENTATION

BASES

3/4.3.3.4 DELETED

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

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

3/4.3.3.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 DELETED

3/4.3.3.8 DELETED

INSTRUMENTATION

BASES

3/4.3.3.9 DELETED |

3/4.3.3.10 DELETED

3/4.3.3.11 DELETED

3/4.3.4 DELETED |

INSTRUMENTATION

BASES

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

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 times.

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 System isolates, (4) the emergency diesel generators start, (5) containment spray pumps start and automatic valves position, (6) containment isolates, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, (11) essential service water pumps start and automatic valves position, and (12) isolate normal control room ventilation and start Emergency Ventilation System.

Engineered Safety Features Actuation System Interlocks

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressure P-11 automatically reinstates safety injection actuation on low pressurizer pressure and low steamline pressure and automatically blocks steamline isolation on negative steamline pressure rate. On decreasing pressure; P-11 allows the manual block of Safety Injection on low pressurizer pressure and low steamline pressure and allows steamline isolation on negative steamline pressure rate to become active upon manual block of low steamline pressure SI.

INSTRUMENTATION

BASES

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated ACTION will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations 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 and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Control Room Emergency Ventilation Systems.

3/4.3.3.2 DELETED

3/4.3.3.3 DELETED

INSTRUMENTATION

BASES

3/4.3.3.4 DELETED

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

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

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 DELETED

3/4.3.3.8 DELETED

INSTRUMENTATION

BASES

3/4.3.3.9 DELETED |

3/4.3.3.10 DELETED

3/4.3.3.11 DELETED

3/4.3.4 DELETED |

3/4.4 REACTOR COOLANT SYSTEM

BASES

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 DNBR above the safety analysis limit DNBR (1.32) during all normal operations and anticipated transients. 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 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat even in the event of a bank withdrawal accident; however, single failure considerations require that three loops be OPERABLE. A single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented; i.e., by opening the Reactor Trip System breakers.

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

In MODE 5 with reactor coolant loops not filled, a single RHR 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 RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR 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.

Addition of borated water with a concentration greater than or equal to the minimum required RWST concentration but less than the actual RCS boron concentration shall not be considered a reduction in boron concentration.

The restrictions on starting a reactor coolant pump in MODES 4 and 5 are provided to prevent RCS 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 RCS 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 RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS 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.

REACTOR COOLANT SYSTEM

BASES

SAFETY VALVES (Continued)

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip and also assuming no operation of the power-operated relief valves or steam dump valves.

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

Addition to the RCS of borated water with a concentration greater than or equal to the minimum required RWST concentration shall not be considered a positive reactivity change. Cooldown of the RCS for restoration of operability of a pressurizer code safety valve, with a negative moderator temperature coefficient, shall not be considered a positive reactivity change provided the RCS is borated to the COLD SHUTDOWN, xenon-free conditions per specification 3.1.1.1.

3/4.4.3 PRESSURIZER

The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The PORVs are equipped with automatic actuation circuitry and manual control capability. Because no credit for automatic PORV operation is taken in the USAR analyses for MODE 1, 2 and 3 transients, the PORVs are considered OPERABLE in either the manual or automatic mode. The automatic mode is the preferred configuration, as this provides pressure relieving capability without reliance on operator action.

REACTOR COOLANT SYSTEM

BASES

OPERATIONAL LEAKAGE (Continued)

The CONTROLLED LEAKAGE limitation restricts operation when the total flow from the reactor coolant pump seals exceeds 8 gpm per RC pump at a nominal RCS pressure of 2235 psig. This limitation ensures adequate performance of the RC pump seals.

The 1 gpm leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since those valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 DELETED

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an

REACTOR COOLANT SYSTEM

BASES

SPECIFIC ACTIVITY (Continued)

assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Wolf Creek Generating Station, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective ACTION. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:
 - a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

- b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.
2. These limit lines shall be calculated periodically using methods provided below.
3. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the 1972 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT} , at the end of 13.6 effective full power years (EFPY) of service life. The 13.6 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT} ; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT} . Therefore, an adjusted reference temperature, based upon the fluence and copper content and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 13.6 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

Values of ΔRT_{NDT} determined in this manner may be used until the results of the next scheduled capsule from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T , and a length of $3/2T$ is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT} , is used and this includes the radiation-induced shift, ΔRT_{NDT} , corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I , for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR} , for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T - RT_{NDT} + 160)] \quad (1)$$

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

The OPERABILITY of two PORVs, or two RHR suction relief valves, or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 368°F. Either PORV or either RHR suction relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water solid RCS.

In addition to opening RCS vents to meet the requirement of Specification 3.4.9.3c., it is acceptable to remove a pressurizer Code safety valve, open a PORV block valve and remove power from the valve operator in conjunction with disassembly of a PORV and removal of its internals, or otherwise open the RCS.

REACTOR COOLANT SYSTEM

BASES

COLD OVERPRESSURE

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for: (1) process and instrumentation uncertainties; and (2) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of both Safety Injection pumps and all but one centrifugal charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary coolant temperature is more than 50°F above reactor coolant temperature. Exceptions to these requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centrifugal charging pump OPERABLE and no Safety Injection pumps OPERABLE is allowed for up to 4 hours. As shown by analysis LOCAs occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single RHR pump with no credit for accumulator injection. Given the short time duration and the condition of having only one centrifugal charging pump OPERABLE is allowed and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging and Safety Injection pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic Safety Injection actuation signals except Containment Pressure - High are blocked. In normal conditions a single failure of the ESF actuation circuitry will result in the starting of at most one train of Safety Injection (one centrifugal charging pump, and one Safety Injection pump). For temperatures above 325°F, an overpressure event occurring as a result of starting two pumps can be successfully mitigated by operation of both PORVs without exceeding Appendix G limit. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of Safety Injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

Although COMS is required to be OPERABLE when RCS temperature is less than 368°F, operation with all centrifugal charging pumps and both Safety Injection pumps OPERABLE is acceptable when RCS temperature is greater than 350°F. Should an inadvertent Safety Injection occur above 350°F, a single PORV has sufficient capacity to relieve the combined flow rate of all pumps. Above 350°F two RCPs and all pressure safety valves are required to be OPERABLE. Operation of an

REACTOR COOLANT SYSTEM

BASES

COLD OVERPRESSURE (Continued)

RCP eliminates the possibility of a 50°F difference existing between indicated and actual RCS temperature as a result of heat transport effects. Considering instrument uncertainties only, an indicated RCS temperature of 350°F is sufficiently high to allow full RCS pressurization in accordance with Appendix G limitations. Should an overpressure event occur in these conditions, the presurizer safety valves provide acceptable and redundant overpressure protection.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H.

3/4.4.10 DELETED

3/4.4.11 DELETED

3/4.6 CONTAINMENT SYSTEMS

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.

Containment leakage rates shall be within the following limits:

- 1) An overall integrated leakage rate of less than or equal to L_a , 0.20% by weight of the containment air per 24 hours at P_a , 48 psig.
- 2) A combined leakage rate of less than $0.60 L_p$ for all penetrations and valves subject to Type B and C tests, when pressurized to P_a , 48 psig.

3/4.6.1.2 DELETED

3/4.6.1.3 CONTAINMENT AIR LOCKS

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

3/4.6.1.4 INTERNAL PRESSURE

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

The maximum peak pressure expected to be obtained from a steam line break event is 48.9 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 50.4 psig, which is less than design pressure and is consistent with the safety analyses.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.5 AIR TEMPERATURE

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

6 DELETED

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

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

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

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit 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.

CONTAINMENT SYSTEMS

BASES

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

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

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

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the 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. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. The educator flow test of 52 gpm with RWST water is equivalent to 40 gpm NaOH solution. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Cooling System ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions. The required design cooling water flow to the Containment Cooling System is verified by the surveillance testing requirements of Specification 4.6.2.3(b) which is performed at 18 month intervals. The testing requirements of Specification 4.6.2.3(a), performed at 31 day intervals, ensure that the fan units and the cooling water flow paths (supply and return) from the Essential Service Water System headers are OPERABLE.

The Containment Cooling System and the Containment Spray System are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time

CONTAINMENT SYSTEMS

BASES

CONTAINMENT COOLING SYSTEM (Continued)

requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC54 thru 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. Operation of the Emergency Exhaust System with the heaters operating 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. These Hydrogen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978.

Adequate mixing of the containment atmosphere following a LOCA is ensured by natural circulation without reliance on a hydrogen mixing systems. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

PLANT SYSTEMS

BASES

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

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

3.4.7.1.6 STEAM GENERATOR ATMOSPHERIC RELIEF VALVES

The operability of the main steamline atmospheric relief valves (ARV's) ensures that reactor decay heat can be dissipated to the atmosphere in the event of a steam generator tube rupture and loss of offsite power and that the Reactor Coolant System can be cooled down for Residual Heat Removal System operation. The number of required ARV's assures that the subcooling can be achieved, consistent with the assumptions used in the steam generator tube rupture analysis, to facilitate equalizing pressures between the Reactor Coolant System and the faulted steam generator. For cooling the plant to RHR initiation conditions, only one ARV is required. In this case, with three ARV's operable, if the single failure of one ARV occurs and another ARV is assumed to be associated with the faulted steam generator, one ARV remains available for required heat removal.

Each ARV is equipped with a manual block valve (in the auxiliary building) to provide a positive shutoff capability should an ARV develop leakage. Closure of the block valves of all ARV's because of excessive seat leakage does not endanger the reactor core; consistent with plant accident and transient analyses, decay heat can be dissipated with the main steamline safety valves or a block valve can be opened manually in the auxiliary building and the ARV can be used to control release of steam to the atmosphere. For the steam generator tube rupture event, primary to secondary leakage can be terminated by depressurizing the Reactor Coolant System with the pressurizer power operated relief valves.

3/4.7.1.7 MAIN FEEDWATER ISOLATION VALVES

The OPERABILITY of the main feedwater isolation valves: (1) provides a pressure boundary to permit auxiliary feedwater addition in the event of a main steam or feedwater line break; (2) limits the RCS cooldown and mass and energy releases for secondary line breaks inside containment; and (3) mitigates steam generator overfill events such as a feedwater malfunction, with protection provided by feedwater isolation via the steam generator high-high level trip signal. The OPERABILITY of the main feedwater isolation valves within the closure times of the surveillance requirements is consistent with the assumptions used in the safety analysis.

3/4.7.2 DELETED

PLANT SYSTEMS

BASES

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses. Each independent CCW loop contains two 100% capacity pumps and, therefore, the failure of one pump does not affect the OPERABILITY of that loop.

3/4.7.4 ESSENTIAL SERVICE WATER SYSTEM

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

3/4.7.5 ULTIMATE HEAT SINK

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

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

3/4.7.6 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the Control Room Emergency Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. 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 charcoal adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1975 and N510-1980 will be used as procedural guides for surveillance testing. Surveillance testing provides assurance that system and component performances continue to be in accordance with performance specifications for Wolf Creek Unit 1, including applicable parts of ANSI N509-1976.

PLANT SYSTEMS

BASES

3/4.7.7 EMERGENCY EXHAUST SYSTEM - AUXILIARY BUILDING

The OPERABILITY of the Emergency Exhaust System ensures that radioactive materials leaking from the ECCS equipment within the Auxiliary Building following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters operating to maintain low humidity for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the charcoal adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-1975 and N510-1980 will be used as procedural guides for surveillance testing. The surveillance requirements associated with the HEPA filters, charcoal adsorbers and heaters are stated in 4.9.13.

3/4.7.8 DELETED

3/4.7.9 DELETED

3/4.7.10 DELETED

3/4.7.11 DELETED

3/4.7.12 DELETED

ELECTRICAL POWER SYSTEMS

BASES

3/4.8.4 DELETED

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3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 is sufficient to prevent reactor criticality during refueling operations. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the Reactor Coolant System. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

When determining compliance with action statement requirements, addition to the RCS of borated water with a concentration greater than or equal to the minimum required RWST concentration shall not be considered to be a positive reactivity change.

3.4.9.3 DECAY TIME

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

3.4.9.4 CONTAINMENT BUILDING PENETRATIONS

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

The OPERABILITY of this system ensures the containment purge penetrations will be automatically isolated upon detection of high radiation levels within containment. The OPERABILITY of this system is required to restrict the release of radioactive materials from the containment atmosphere to the environment.

Equivalent isolation methods for the emergency personnel escape lock and containment wall penetrations ensure releases from containment are prevented for credible accident scenarios. The isolation techniques must be approved by an engineering evaluation and may include use of a material that can provide a temporary, pressure tight seal capable of maintaining the integrity of the penetrations and airlock to restrict the release of radioactive material from a fuel element rupture.

REFUELING OPERATIONS

BASES

3/4.9.5 DELETED

3/4.9.6 DELETED

3/4.9.7 DELETED

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification. The minimum of 1000 gpm allows flow rates which provide additional margin against vortexing at the RHR pump suction while in a reduced RCS inventory condition.

Addition of borated water with a concentration greater than or equal to the minimum required RWST concentration but less than the actual RCS boron concentration shall not be considered a reduction in boron concentration.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of RHR capability. With the reactor vessel head removed and at least 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT VENTILATION SYSTEM

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

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

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

3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

The restrictions placed on spent fuel assemblies stored in Region 2 of the spent fuel pool ensure inadvertent criticality will not occur.

REFUELING OPERATIONS

BASES

3/4.9.13 EMERGENCY EXHAUST SYSTEM - FUEL BUILDING

The limitations on the Emergency Exhaust System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating to maintain low humidity 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. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1975 and N510-1980 will be used as procedural guides for surveillance testing.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 DELETED

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

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

3/4.10.3 PHYSICS TESTS

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

3/4.10.4 REACTOR COOLANT LOOPS

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

3/4.10.5 DELETED

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 DELETED

3/4.11.1.2 DELETED

3/4.11.1.3 DELETED

3/4.11.1.4 DELETED

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DELETED

3/4.11.2.2 DELETED

3/4.11.2.3 DELETED

3/4.11.2.4 DELETED

3/4.11.2.5 DELETED

3/4.11.2.6 DELETED

3/4.11.3 DELETED

3/4.11.4 DELETED

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

e. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- 1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with methodology in the ODCM,
- 2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,
- 3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,
- 4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,
- 5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,
- 6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,
- 7) Limitations of the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

- 8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,
- 9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,
- 10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

- 1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.
- 2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and the modifications to the monitoring program are made if required by the results of this census, and
- 3) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

6.8.5 a. Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

1. The limits for concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM and a surveillance program to ensure the limits are maintained.
2. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an UNRESTRICTED AREA in the event of an uncontrolled release of the tanks'

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contents, consistent with Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases due to Waste Gas System Leak or Failure."

3. A surveillance program to ensure that the quantity of radioactivity contained in following outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste system, is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA, in the event of an uncontrolled release of the tank's contents.
 - a. Reactor Makeup Water Storage Tank.
 - b. Refueling Water Storage Tank,
 - c. Condensate Storage Tank, and
 - d. Outside Temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

- b. Reactor Coolant Pump Flywheel Inspection Program

Each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, dated August 1975.

- c. Containment Tendon Surveillance Program

This program provides controls for monitoring tendon performance, including the effectiveness of the tendon corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial plant operation as well as periodic testing thereafter. The Containment Tendon Surveillance Program, and its inspection frequencies and acceptance criteria, shall be in accordance with Wolf Creek Generating Station position on draft Revision 3 of Regulatory Guide 1.35 dated April 1989.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Containment Tendon Surveillance Program inspection frequencies.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the NRC Regional Office unless otherwise noted.