ATTACHMENT IV

PROPOSED TECHNICAL SPECIFICATION CHANGES

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3/4.9 REFUELING OPERATIONS

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,

e ~~d~~. The containment leakage rates are within the limits listed in the Bases of Specification 3.6.1.~~2~~1, and

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

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% $\Delta k/k$ ~~for four loop operation~~.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.3% $\Delta k/k$, ~~immediately~~ 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 equiv ent 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. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

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

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

*See Special Test Exception Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

b ~~c.~~ ~~When in MODE 3, 4, or 5,~~ 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.

~~4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e above. 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.~~

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

DELETE

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System if the Boric Acid Storage System is OPERABLE as given in Specification 3.1.2.5a. for MODES 5 and 6 or as given in Specification 3.1.2.6a. for MODE 4; or

b. The flow path from the refueling water storage tank via a centrifugal charging pump to the Reactor Coolant System if the refueling water storage tank is OPERABLE as given in Specification 3.1.2.5b. for MODES 5 and 6 or as given in Specification 3.1.2.6b. for MODE 4.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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 ±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 4.1.1.1.1b. 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 4.1.1.1.1b, with the control banks at the maximum insertion limit of Specification 3.1.3.6.

Delete

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System, and

b. Two flow paths from the refueling water storage tank via centrifugal charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.3% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

c. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the Reactor Coolant System.

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

DELETE

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One centrifugal charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

With no centrifugal charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required centrifugal charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, that the pump develops a differential pressure of greater than or equal to 2400 psid when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All centrifugal charging pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable* at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge of 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.

CHARGING PUMPS - OPERATING

DELETE

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two centrifugal charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

BORATED WATER SOURCE - SHUTDOWN

DELETE

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 2968 gallons,

2) Between 7000 and 7700 ppm of boron, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 55,416 gallons,

2) A minimum boron concentration of 2400 ppm, and

3) A minimum solution temperature of 37°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 37°F.

BORATED WATER SOURCES - OPERATING

DELETE

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water sources shall be OPERABLE as required by Specification 3.1.2.2 for MODES 1, 2, and 3 and one of the following borated water sources shall be OPERABLE as required by Specification 3.1.2.1 for MODE 4:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 17,658 gallons,

2) Between 7000 and 7700 ppm of boron, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 394,000 gallons,

2) Between 2400 and 2500 ppm of boron,

3) A minimum solution temperature of 37°F, and

4) A maximum solution temperature of 100°F.

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

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources in MODE 1, 2 or 3, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1.3% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable in MODE 1, 2, or 3, 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.

c. With no borated water source OPERABLE in MODE 4, restore one borated water source to OPERABLE status within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

Delete

4.1.2.6 Each required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated 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.

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	ACTION 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)

USCAT 1-14

ACTION 1 - ~~Determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.~~

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.

INSERT 1-14

ACTION 1 - 1. Determine that the SHUTDOWN MARGIN is greater than or equal to 1.3% Δk/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 (Continued)

is greater than or equal to 1.3% ΔK/K

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

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

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

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

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

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 INSERT 1-15

INSERT 1-15

4.1.3.1.3 Prior to reactor criticality, verify that the rod drop time of the full-length shutdown and control rods is in accordance with USAR Section 16.1.3.2, with $T_{AVG} \geq 551°F$, 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.

POSITION INDICATION SYSTEM-SHUTDOWN

Delete

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*With the Reactor Trip System breakers in the closed position.

#See Special Test Exception Specification 3.10.5.

DELETE

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the physical fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the rod drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

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, and

c. At least once per 18 months.

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:

INSERT 1-21A

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

c ~~b~~. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

d ~~c~~. 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 INSERT 1-21B

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

INSERT 1-21A

a. Within 1 hour, verify that the SHUTDOWN MARGIN is greater than or equal to 1.3% Δk/k or initiate boration until the SHUTDOWN MARGIN is restored to greater than or equal to 1.3% Δk/k, and

INSERT 1-21B

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.

MOVABLE INCORE DETECTORS

DELETE

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$.

ACTION:

a. With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions.

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$.

SEISMIC INSTRUMENTATION

DELETE

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

DELETE

TABLE 3.3-7

SEISMIC MONIT RING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE			MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Peak Recording Accelerographs				
a. Radwaste Base Slab	± 1.0g			1
b. Control Room	± 1.0g			1
c. ESW Pump Facility	± 1.0g			1
d. Ctmt Structure	± 2.0g			1
e. Auxiliary Bldg. SI Pump Suctions	± 1.0g			1
f. SGB Piping	± 5.0g			1
g. SGC Support	± 1.0g			1
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)				
a. Ctmt. Base Slab	± 1.0g			1
b. Ctmt. Oper. Floor	± 1.0g			1
c. Reactor Support	± 1.0g			1
d. Aux. Bldg. Base Slab	± 1.0g			1
e. Aux. Bldg. Control Room Air Filter	± 1.0g			1
f. Free Field	± 0.5g			1
3. Triaxial Response-Spectrum Recorder (Passive)				
Ctmt. Base Slab	± 1.0g			1
4. Triaxial Seismic Switches	ACCELERATION LEVEL			
	North	East	Vertical	
a. OBE Ctmt. Base Slab	0.06g	0.06g	0.06g	1
b. SSE Ctmt. Base Slab	0.15g	0.15g	0.16g	1
c. OBE Ctmt. Oper. Fl.	0.07g	0.07g	0.07g	1
d. SSE Ctmt. Oper. Fl.	0.16g	0.17g	0.16g	1
e. System Trigger	0.01g	0.01g	0.01g	1

DELETE

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Peak Recording Accelerographs			
a. Radwaste Base Slab	N.A.	R	N.A.
b. Control Room	N.A.	R	N.A.
c. ESW Pump Facility	N.A.	R	N.A.
d. Ctmt Structure	N.A.	R	N.A.
e. Auxiliary Bldg. SI Pump Suction	N.A.	R	N.A.
f. SGB Piping	N.A.	R	N.A.
g. SGC Support	N.A.	R	N.A.
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)			
a. Ctmt. Base Slab	M	R	SA
b. Ctmt. Oper. Floor	M	R	SA
c. Reactor Support	M	R	SA**
d. Aux. Bldg. Base Slab	M	R	SA**
e. Aux. Bldg. Control Room Filters	M	R	SA**
f. Free Field	M	R	SA**
3. Triaxial Response-Spectrum Recorder (Passive)			
Ctmt. Base Slab	N.A.	R	N.A.*
4. Triaxial Seismic Switches			
a. OBE Ctmt. Base Slab	M	R	SA
b. SSE Ctmt. Base Slab	M	R	SA
c. OBE Ctmt. Oper. Fl.	M	R	SA
d. SSE Ctmt. Oper. Fl.	M	R	SA
e. System Trigger	M	R	SA

*Checking at the Main Control Board Annunciation for contact closure output in the Control Room shall be performed at least once per 184 days.
**The Bi-stable Trip Setpoint need not be determined during the performance of an ANALOG CHANNEL OPERATIONAL TEST.

METEOROLOGICAL INSTRUMENTATION

DELETE

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-5.

Delete

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. Wind Speed	Nominal Elev. 10m	1
	Nominal Elev. 60m	1
2. Wind Direction	Nominal Elev. 10m	1
	Nominal Elev. 60m	1
3. Air Temperature - ΔT	Nominal Elev. 10m-60m	1

DELETE

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Nominal Elev. 10m	D	SA
b. Nominal Elev. 60m	D	SA
2. Wind Direction		
a. Nominal Elev. 10m	D	SA
b. Nominal Elev. 60m	D	SA
3. Air Temperature - ΔT		
a. Nominal Elev. 10-60m	D	SA

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:

DELETE

INSERT 3-53

~~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 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 number of OPERABLE accident monitoring instrumentation channels, except the containment radiation level monitor and the unit vent - high range noble gas monitor, less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours; 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 the containment radiation level monitor or the unit vent - high range noble gas monitor 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 the inoperable channel to OPERABLE status within 7 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days that provides actions taken, cause of the inoperability and plans and schedule for restoring the channels to OPERABLE status.~~

e ~~d~~. 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.

INSERT 3-53

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.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

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

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

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

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure – 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. ~~PORV Position Indicator*~~ Deleted	~~M~~	~~N.A.~~
13. ~~PORV Block Valve Position Indicator**~~ Deleted	~~M~~	~~N.A.~~
14. ~~Safety Valve Position Indicator~~ Neutron Flux	M	~~N.A.~~ 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. ~~Unit Vent - High Range Noble Gas Monitor~~ Reactor Vessel Level Indicating System	M	R

~~*Not applicable if the associated block valve is in the closed pos~~ition.

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

(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.

(1) Neutron Detectors may be excluded from Channel Calibration.

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

Delete

3.3.3.9 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACITON:

a. With one or more Loose-Part Detection System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each channel of the Loose-Part Detection System shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST except for verification of Setpoint at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

INSTRUMENTATION

Delete

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The explosive gas monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.5 are not exceeded.

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable and take the ACTION shown in Table 3.3-13.

b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within 30 days, and, if unsuccessful, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 to explain why this inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

TABLE 3.3-13

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. (Not Used)			
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System			
a. Hydrogen Monitor	1/Recombiner	**	44
b. Oxygen Monitor	2/Recombiner	**	42

Delete

Delete

TABLE 3.3-13 (Continued)

** During WASTE GAS HOLDUP SYSTEM operation.

ACTION STATEMENTS

ACTION 38 - (Not Used)

ACTION 39 - (Not Used)

ACTION 40 - (Not Used)

ACTION 41 - (Not Used)

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

ACTION 43 - (Not Used)

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

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. (Not used)				
2. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System				
a. Inlet Hydrogen Monitor	D	Q(4)	M	**
b. Outlet Hydrogen Monitor	D	Q(4)	M	**
c. Inlet Oxygen Monitor	D	Q(5)	M	**
d. Outlet Oxygen Monitor	D	Q(6)	M	**

Delete

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

Delete

** During WASTE GAS HOLDUP SYSTEM operation.

(1) (Not Used)

(2) (Not Used)

(3) (Not Used)

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

a. One volume percent hydrogen, balance nitrogen and

b. Four volume percent hydrogen, balance nitrogen.

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

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

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

a. 10 ppm by volume oxygen, balance nitrogen, and

b. 80 ppm by volume oxygen, balance nitrogen.

INSTRUMENTATION

Delete

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

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

ACTION:

a. With one stop valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lines or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position:

1) Four high pressure turbine stop valves,
2) Six low pressure turbine reheat stop valves, and
3) Six low pressure turbine reheat intercept valves.

b. At least once per 31 days by cycling each of the four high pressure main turbine governor valves through at least one complete cycle from the running position;

c. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position;

d. At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems; and

e. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

---

*Not applicable in MODE 2 or 3 with all main steam line isolation valves and associated bypass valves in the the closed position and all other steam flow paths to the turbine isolated.

3/4.4.2 SAFETY VALVES

SHUTDOWN

Delete

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 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.

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

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

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

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one or both block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to OPERABLE status within 72 hours; 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

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

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

INSERT 4-10 →

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

INSERT 4-10

4.4.4.3 Both PORV position indicators shall be demonstrated OPERABLE at least once per 31 days by performance of a CHANNEL CHECK unless the associated block valve is in the closed position.

4.4.4.4 Both PORV block valve position indicators shall be demonstrated OPERABLE at least once per 31 days by performance of a CHANNEL CHECK unless the block valve is verified in the closed position and power is removed.

3/4.4.5 STEAM GENERATORS

Delete

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

Delete

1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

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

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

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

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

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

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

Delete

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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

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

   4) A main steam line or feedwater line break.

Delete

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

Delete

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

TABLE NOTATIONS

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

Delete

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N. A.	N. A.	N. A.	N. A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N. A.	N. A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N. A.	N. A.
	C-3	Inspect all tubes in this S. G., plug defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	All other S. G.s are C-1	None	N. A.	N. A.
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N. A.	N. A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(2) of 10 CFR Part 50	N. A.	N. A.

$S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

Delete

REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

Delete

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

Delete

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

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

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

Delete

TABLE ·.4-3

REACTOR COOLAN: SYSTEM

CHEMISTRY SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least once per 72 hours
Chloride	At least once per 72 hours
Fluoride	At least once per 72 hours

*Not required with T_{avg} less than or equal to 250°F

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

Delete

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 583°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

Delete

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

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

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Delete

LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor vessel head vent path consisting of at least two valves in series powered from emergency busses shall be OPERABLE and closed.

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

ACTION:

With the above reactor vessel head vent path inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING, and

c. Verifying flow through the reactor vessel head vent paths during venting during COLD SHUTDOWN or REFUELING.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 ECCS SUBSYSTEMS - $T_{AVG} \leq 200°F$

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 ~~with the water level above the top of the Reactor Vessel flange,~~ and Mode 6 with the Reactor Vessel head on.* ~~and with the water level above the top of the Reactor Vessel flange.~~

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.

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; and

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 4.6.1.2d.~~ for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.
USAR Section 16.6.1.1

*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.

Delete

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

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

2) Less than or equal to L_t, 0.020% by weight of the containment air per 24 hours at P_t, 24 psig.

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

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

ACTION:

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

b. If the Reactor Coolant System temperature is above 200 degrees F, with the measured combined leakage rate for all penetrations and valves subject to Types B and C test exceeding 0.60 L_a,

1) Restore the combined leakage rate to less than 0.60 L_a within 4 hours by one of the following methods:

a) Repairing the failed containment isolation component, or

b) Isolating the penetration containing the failed component by closing and the deactivating one automatic valve, or

c) Isolating the penetration containing the failed component by closing one manual valve, or

d) Isolating the penetration containing the failed component by using a blind flange.

2) If the combined leakage rate is not restored to less than 0.60 L_a within 4 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Delete

SURVEILLANCE REQUIREMENTS

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

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

SURVEILLANCE REQUIREMENTS (Continued)

Delete

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t at which time the above test schedule may be resumed;

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

1) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a or 0.25 L_t;

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

3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a or 0.75 L_t and 1.25 L_t.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 48 psig, at intervals no greater than 24 months except for tests involving:

1) Air locks,

2) Purge supply and exhaust isolation valves with resilient material seals, and

3) Valves pressurized with fluid from a seal system.

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.7.2 and 4.6.1.7.4, as applicable;

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (53 psig), and the seal system capacity is adequate to maintain system pressure for at least 30 days; and

h. The provisions of Specification 4.0.2 are not applicable.

Delete

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

a. With the abnormal degradation indicated by the conditions in Specification 4.6.1.6.1a.4, restore the tendons to the required level of integrity or verify that containment integrity is maintained within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the indicated abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Specification 4.6.1.6, restore the containment vessel to the required level of integrity or verify that containment integrity is maintained within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Vessel Tendons. The structural integrity of the prestressing tendons of the containment vessel shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The structural integrity of the tendons shall be demonstrated by:

a. Determining that a random but representative sample of at least 11 tendons (4 inverted U and 7 hoop) each have an observed lift-off force within the predicted limits established for each tendon. For each subsequent inspection one tendon from each group (1 inverted U and 1 hoop) shall be kept unchanged to develop a history and to correlate the observed data. The procedure of inspection and the tendon acceptance criteria shall be as follows:

SURVEILLANCE REQUIREMENTS (Continued)

Delete

1. If the measured prestressing force of the selected tendon in a group lies above the prescribed lower limit, the lift-off test is considered to be a positive indication of the sample tendon's acceptability,

2. If the measured prestressing force of the selected tendon in a group lies between the prescribed lower limit and 90% of the prescribed lower limit, two adjacent (accessible) tendons, one on each side of this tendon shall be checked for their prestressing forces. If the prestressing forces of these two tendons are above 95% of the prescribed lower limits for the tendons, all three tendons shall be restored to the required level of integrity, and the tendon group shall be considered as acceptable. If the measured prestressing force of any two tendons falls below 95% of the prescribed lower limits of the tendons, additional lift-off testing shall be done to detect the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the containment structure,

3. If the measured prestressing force of any tendon lies below 90% of the prescribed lower limit, the defective tendon shall be completely detensioned and additional lift-off testing shall be done so as to determine the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the containment structure,

4. If the average of all measured prestressing forces for each group (corrected for average condition) is found to be less than the minimum required prestress level at the anchorage locations for that group, the condition shall be considered as abnormal degradation of the containment structure,

5. If from consecutive surveillances the measured prestressing forces for the same tendon or tendons in a group indicate a trend of prestress loss larger than expected and the resulting prestressing forces will be less than the minimum required for the group before the next scheduled surveillance, additional lift-off testing shall be done so as to determine the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the containment structure, and

6. Unless there is abnormal degradation of the containment vessel during the first three inspections, the sample population for subsequent inspections shall include at least 6 tendons (3 hoop, 3 inverted U).

Delete

SURVEILLANCE REQUIREMENTS (Continued)

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group. A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire sample (which shall include the broken wire if so identified) that:

   1. The tendon wires are free of corrosion, cracks, and damage, and

   2. A minimum tensile strength of 240 ksi (guaranteed ultimate strength of the tendon material) exists for at least three wire samples (one from each end and one at mid-length) cut from each removed wire.

   Failure to meet the requirements of 4.6.1.6.1.b shall be considered as an indication of abnormal degradation of the containment structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to at least the force level recorded prior to detensioning or the predicted value, whichever is greater, with the tolerance within minus zero to plus 6%, but not to exceed 70% of the guaranteed ultimate tensile strength of the tendons. During retensioning of these tendons the changes in load and elongation shall be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 10% from that recorded during the installation, an investigation shall be made to ensure that the difference is not related to wire failures or slip of wires in anchorages. This condition shall be considered as an indication of abnormal degradation of the containment structure.

d. Verifying the OPERABILITY of the sheathing filler grease by assuring:

   1. There are no changes in the presence or physical appearance of the sheathing filler-grease including the presence of free water,

   2. Amount of grease replaced does not exceed 5% of the net duct volume, when injected at ± 10% of the specified installation pressure,

   3. Minimum grease coverage exists for the different parts of the anchorage system,

   4. During general visual examination of the containment exter surface, that grease leakage that could affect containment integrity is not present, and

Delete

SURVEILLANCE REQUIREMENTS (Continued)

5. The chemical properties of the filler material are within the tolerance limits specified as follows:

Water Content	0 - 10% by dry weight
Chlorides	0 - 10 ppm
Nitrates	0 - 10 ppm
Sulfides	0 - 10 ppm
Reserved Alkalinity	>0

Failure to meet the requirements of 4.6.1.6.1.d shall be considered as an indication of abnormal degradation of the containment structure.

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. As an assurance of the structural integrity of the containment vessel, tendon anchorage assembly hardware (such as bearing plates, stressing washers, wedges, and buttonheads) of all tendons selected for inspection shall be visually examined. Tendon anchorages selected for inspection shall be visually examined to the extent practical without dismantling the load bearing components of the anchorages. Bottom grease caps of all vertical tendons shall be visually inspected to detect grease leakage or grease cap deformations. The surrounding concrete shall also be checked visually for indication of any abnormal condition. The frequency of this surveillance shall be in accordance with 4.6.1.6.1. Significant grease leakage, grease cap deformation or abnormal concrete condition shall be considered as an indication of abnormal degradation of the containment structure.

4.6.1.6.3 Containment Vessel Surfaces. The exterior surface of the containment shall be visually examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 sq. ft. or more, other surface deterioration or disintegration, or grease leakage, each of which shall be considered as evidence of abnormal degradation of structural integrity of the containment. This inspection shall be performed prior to the Type A containment leakage rate test.

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 4.6.1.2d.~~ USAR SECTION 16.6.1.1, 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.

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.

~~*For valves with excessive leakage, refer to Technical Specification 3.6.1.2.~~

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
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 and Specification 3.3.2 apply to the Main Steam Isolation Valves and Main Feedwater Isolation Valves. ~~respectively.~~

3.7.1.7

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

Delete

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

b. With both hydrogen analyzers inoperable, restore at least one analyzer to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each containment hydrogen analyzer shall be demonstrated OPERABLE by the performance of an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 31 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing ten volume percent hydrogen, balance nitrogen.

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

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

Delete

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generator shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.

Delete

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.8g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-2. The visual inspection interval for each type snubber shall be determined based upon the criteria provided in Table 4.7-2 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment 44.

SURVEILLANCE REQUIREMENTS (Continued)

Delete

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, and (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; or (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.8f. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to determine system operability with an unacceptable snubber. If operability cannot be justified, the system shall be declared inoperable and the ACTION requirements shall be met.

d. Transient Event Inspections

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

Delete

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers of each type shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.8f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.8f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on the Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$ where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

Delete

e. Functional Tests (Continued).

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed rate, or release rate where required, is present in both tension and compression, within the specified range; and

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Service Life Monitoring Program

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

SURVEILLANCE REQUIREMENTS (Continued)

Delete

g. Service Life Monitoring Program (Continued)

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.8e. for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

Delete

TABLE 4.7-2
SNUBBER VISUAL INSPECTION INTERVAL

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population per Catergory (Notes 1 and 2)	Column A Extend Interval (Extend 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber category shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, categories must be determined and documented before any inspection and that determination shall be the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population per category and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, and C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Delete

Table 4.7-2 (Continued)
SNUBBER VISUAL INSPECTION INTERVAL

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Column B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

Delete

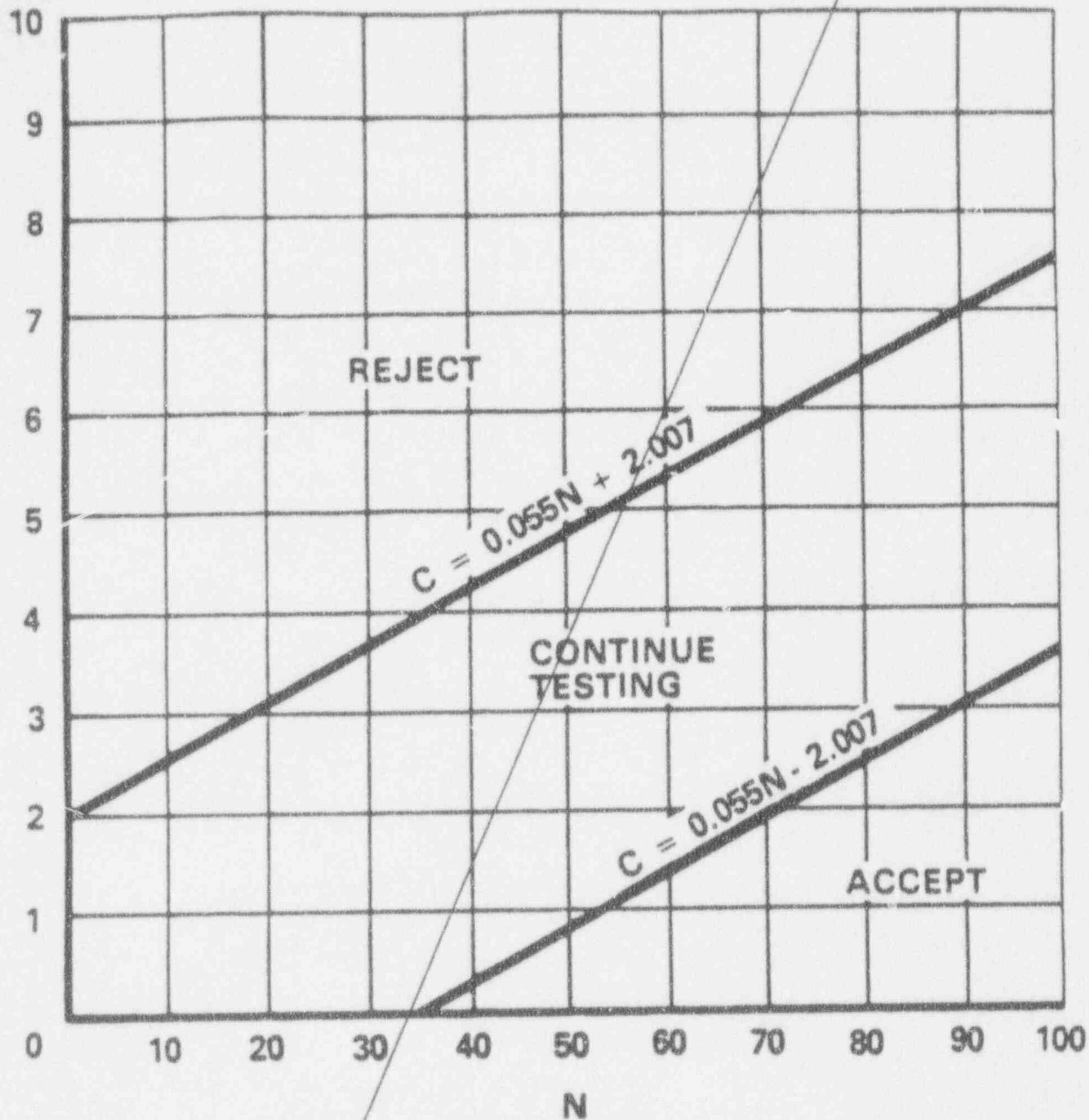

FIGURE 4.7-1

SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

Delete

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma-emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

4.7.9.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

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

   2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

Delete

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

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

4.7.9.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

3/4.7.12 AREA TEMPERATURE MONITORING

Delete

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature limit of each area given in Table 3.7-4 shall not be exceeded for more than 8 hours or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-4 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-4 by more than 30°F, prepare and submit a Special Report as required by ACTION a. above, and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.12 The temperature in each of the areas shown in Table 3.7-4 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7-4

AREA TEMPERATURE MONITORING

Delete

	AREA	MAXIMUM TEMPERATURE LIMIT (°F)
1.	ESW Pump Room A	119
2.	ESW Pump Room B	119
3.	Auxiliary Feedwater Pump Room A	119
4.	Auxiliary Feedwater Pump Room B	119
5.	Turbine Driven Auxiliary Feedwater Pump Room	147
6.	ESF Switchgear Room I	87
7.	ESF Switchgear Room II	87
8.	RHR Pump Room A	119
9.	RHR Pump Room B	119
10.	CTMT Spray Pump Room A	119
11.	CTMT Spray Pump Room B	119
12.	Safety Injection Pump Room A	119
13.	Safety Injection Pump Room B	119
14.	Centrifugal Charging Pump Room A	119
15.	Centrifugal Charging Pump Room B	119
16.	Electrical Penetration Room A	101
17.	Electrical Penetration Room B	101
18.	Component Cooling Water Room A	119
19.	Component Cooling Water Room B	119
20.	Diesel Generator Room A	119
21.	Diesel Generator Room B	119
22.	Control Room	84

Delete

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 For each containment penetration provided with a penetration conductor overcurrent protective device(s), each device shall be OPERABLE.

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

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 Protective devices required to be OPERABLE as containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE.

a. At least once per 18 months:

   1) By verifying that the 13.8 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers, and performing the following:

      a) A CHANNEL CALIBRATION of the associated protective relays,

      b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed, and

Delete

SURVEILLANCE REQUIREMENTS (Continued)

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal Setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

Delete

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

Delete

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The refueling machine used for movement of fuel assemblies having:

1) A minimum capacity of 4800 pounds,

2) Automatic overload cutoffs with the following Setpoints:

a) Primary - less than or equal to 250 pounds above the indicated suspended weight for wet conditions and less than or equal to 350 pounds above the indicated suspended weight for dry conditions, and

b) Secondary - less than or equal to 150 pounds above the primary overload cutoff.

3) An automatic load reduction trip with a Setpoint of less than or equal to 250 pounds below the suspended weight for wet conditions or dry conditions.

b. The auxiliary hoist used for latching and unlatching drive rods and thimble plug handling operations having:

1) A minimum capacity of 3000 pounds, and

2) A 1000-pound load indicator which shall be used to monitor lifting loads for these operation.

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

ACTION:

With the requirements for refueling machine and/or auxiliary hoist OPERABILITY not satisified, suspend use of any inoperable refueling machine crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 The refueling machine used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior

SURVEILLANCE REQUIREMENTS (Continued)

to the movement of fuel assemblies in the reactor vessel by performing a load test of at least 125% of the secondary automatic overload cutoff and demonstrating an automatic load cutoff when the refueling machine load exceeds the Setpoints of Specification 3.9.6a.2) and by demonstrating an automatic load reduction trip when the load reduction exceeds the Setpoint of Specification 3.9.6a.3).

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the movement of drive rods within the reactor vessel by performing a load test of at least 1250 pounds.

Delete

Delete

REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2250 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage facility.

APPLICABILITY: With fuel assemblies in the spent fuel storage facility.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

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

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2250 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

WATER LEVEL - REACTOR VESSEL

CONTROL RODS

LIMITING CONDITION FOR OPERATION

Delete

3.9.10.2 At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor pressure vessel.

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

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of control rods within the pressure vessel.

SURVEILLANCE REQUIREMENTS

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

Delete

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

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

Delete

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided only one shutdown or control bank is withdrawn from the fully inserted position at a time.

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements and during surveillance of digital rod position indicators for OPERABILITY.

ACTION:

With the Position Indication System inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

Delete

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 150 Curies, excluding tritium and dissolved or entrained noble gases.

a. Reactor Makeup Water Storage Tank,
b. Refueling Water Storage Tank,
c. Condensate Storage Tank, and
d. Outside temporary tanks, excluding demineralizer vessels and liners being used to solidify radioactive wastes.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added and within 7 days following any addition of radioactive material to the tank.

Delete

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 3% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 3% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a. above.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

Delete

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 2.5×10^5 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, and within 48 hours, reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days when radioactive materials are being added and within 7 days following any addition of radioactive material to the tank.

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.

INSERT 6.8.5 →

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.

INSERT 6.8.5

The following programs, relocated from the Technical Specifications to USAR Chapter 16, shall be implemented and maintained:

a. Explosive Gas and Storage Tank Radioactivity Monitoring Program
b. Turbine Overspeed Protection Reliability Program
c. Steam Generator Tube Surveillance Program
d. Reactor Coolant Pump Flywheel Inspection Program
e. Snubber Inspection Program
f. Area Temperature Monitoring Program
g. Primary Water Chemistry Program
h. Containment Tendon Surveillance Program

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 it 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_{NDT} temperature.

INSERT B1

3/4.1.2 ~~BORATION SYSTEMS~~ DELETED

~~The Boration Systems ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) centrifugal charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.~~

~~With the RCS average temperature equal to or greater than 350°F a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% Δk/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 17,658 gallons of 7000 ppm borated water from the boric acid storage tanks or 83,754 gallons of 2400 ppm borated water from the RWST. With the RCS average temperature less than 350°, only one boron injection flow path is required.~~

BORATION SYSTEMS (Continued)

With the RCS temperature below 200°F, one Boration System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable in MODES 4, 5, and 6 provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or an RHR suction relief valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.3% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2968 gallons of 7000 ppm borated water from the boric acid storage tanks or 14,071 gallons of 2400 ppm borated water from the RWST.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. In the case of the boric acid tanks, all of the contained volume is considered usable. The required usable volume may be contained in either or both of the boric acid tanks.

The limits on contained water volume and boron concentration of the RWST also 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 OPERABILITY of one Boration System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

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.

3/4.1.3 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

INSERT B1

3/4.1.1.5 CORE REACTIVITY

The core is considered to be operating within acceptable design limits when measured core reactivity is within ±1% Δk/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 (±1% Δk/k of the predicted value) ensures plant operation is maintained within the assumptions of the safety analyses.

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.

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.

MOVABLE CONTROL ASSEMBLIES (Continued)

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 ~~of Specification 3.1.3.4~~ during the last performance of Surveillance Requirement ~~4.1.3.4~~. 4.1.3.1.3.

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.

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 ~~MOVABLE INCORE DETECTORS~~ DELETED

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q^M(X,Y,Z)$ or $F_{\Delta H}^M(X,Y)$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Neutron Flux channel is inoperable.

3/4.3.3.3 ~~SEISMIC INSTRUMENTATION~~ DELETED

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

3/4.3.3.4 ~~METEOROLOGICAL INSTRUMENTATION~~ DELETED

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

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN 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

3/4.3.3.9 ~~LOOSE-PART DETECTION INSTRUMENTATION~~ DELETED

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

3/4.3.3.10 DELETED

3/4.3.3.11 DELETED

3/4.3.4 ~~TURBINE OVERSPEED PROTECTION~~ DELETED

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

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. ~~The relief capacity of a single safety valve is adequate to relieve any overpressure~~

SAFETY VALVES (Continued)

condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

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.~~2~~ 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.

3/4.4.5 ~~STEAM GENERATORS~~ DELETED

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

Unscheduled inservice inspections are performed on each steam generator following: 1) reactor to secondary tube leaks; 2) seismic occurrence greater than the Operating Basis Earthquake; 3) a loss-of-coolant accident requiring actuation of the Engineered Safety Features, which for this specification is defined to be a break greater than that equivalent to the severance of a 1" inside diameter pipe, or, for a main steamline or feedline, a break greater than that equivalent to a steam generator safety valve failing open; to ensure that steam generator tubes retain sufficient integrity for continued operation. Transients less severe than these do not require inspections because the resulting stresses are well within the stress criteria established by Regulatory Guide 1.121 which unplugged steam generator tubes must be capable of withstanding.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to

~~STEAM GENERATORS (Continued)~~

~~reliably detect degradation that has penetrated 20% of the original tube wall thickness.~~

~~Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.~~

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

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

3/4.4.6.2 OPERATIONAL LEAKAGE

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

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

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA condtions.

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

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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 ~~CHEMISTRY~~ DELETED

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

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

3/4.4.8 SPECIFIC ACTIVITY

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

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. The ndary side of the steam generator must not be pressurized abc psig if the temperature of the steam generator is below 70°F.~~

~~4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 583°F.~~

3 ~~5~~. System preservice hydrotests and in-service leak and hydrotests s be performed at pressures in accordance with the requirements o ME 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.

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.

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

The OPERABILITY of two PORVs, or 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.

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 pressurizer 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 ~~STRUCTURAL INTEGRITY~~ DELETED

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

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.11 ~~REACTOR COOLANT SYSTEM VENTS~~ DELETED

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of a reactor vessel head vent path ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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.

INSERT B6-1

3/4.6.1.2 ~~CONTAINMENT LEAKAGE~~ DELETED

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

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

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

3/4.6.1.3 CONTAINMENT AIR LOCKS

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

INSERT B6-1

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_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a, 48 psig.

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.

3/4.6.1.5 AIR TEMPERATURE

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

3/4.6.1.6 ~~CONTAINMENT VESSEL STRUCTURAL INTEGRITY~~ DELETED

~~This limitation ensures that the structural integrity of the containment will be maintained in accordance with safety analysis requirements for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 50.4 psig in the event of a steam line break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.~~

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

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

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 ~~of Specification 3.6.1.2.b.~~ shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

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

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

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

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

SPRAY ADDITIVE SYSTEM (Continued)

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 tim 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 ~~detection and~~ 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 Hydro-

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 astmosphere. 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.

INSERT B2 →

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

~~The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.~~

INSERT B2

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.

3/4.7.8 ~~SNUBBERS~~ DELETED

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer. Snubbers may also be classified and grouped by inaccessible or accessible for visual inspection purposes. Therefore, each snubber type may be grouped for inspection in accordance with accessibility.

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

The visual inspection frequency is based upon maintaining a constant level of snubber protection during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection of each type. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing. Since the visual

SNUBBERS (Continued)

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

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

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

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

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

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

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

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.9 ~~SEALED SOURCE CONTAMINATION~~ DELETED

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

SEALED SOURCE CONTAMINATION (Continued)

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 DELETED

3/4.7.11 DELETED

3/4.7.12 ~~AREA TEMPERATURE MONITORING~~ DELETED

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ±3°F.

3/4.8.4 ~~ELECTRICAL EQUIPMENT PROTECTIVE DEVICES~~ DELETED

~~Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.~~

~~The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.~~

~~A list of containment penetration conductor overcurrent protective devices whose circuit limiting fault current exceeds the penetration rating, with information of location and size and equipment powered by the protected circuit, is available at the plant site in accordance with Section 50.71(c) of 10 CFR Part 50. The addition or deletion of any containment penetration conductor overcurrent protective device would be made in accordance with Section 50.59 of 10 CFR Part 50.~~

3/4.9.5 ~~COMMUNICATIONS~~ DELETED

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

3/4.9.6 ~~REFUELING MACHINE~~ DELETED

~~The OPERABILITY requirements for the refueling machine and auxiliary hoist ensure that: (1) manipulator cranes will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.~~

3/4.9.7 ~~CRANE TRAVEL - SPENT FUEL STORAGE FACILITY~~ DELETED

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

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.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 ~~SHUTDOWN MARGIN~~ DELETED

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

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

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

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the 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 ~~POSITION INDICATION SYSTEM-SHUTDOWN~~ DELETED

~~This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements.~~

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 ~~LIQUID HOLDUP TANKS~~ DELETED

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

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 ~~EXPLOSIVE GAS MIXTURE~~ DELETED

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 ~~GAS STORAGE TANKS~~ DELETED

~~The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.~~

3/4.11.3 DELETED

3/4.11.4 DELETED

ATTACHMENT V

RESULTS OF APPLICATION OF THE NRC FINAL POLICY STATEMENT

ON

TECHNICAL SPECIFICATION IMPROVEMENTS

Introduction

The NRC's Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, 58 FR 39132, July 22, 1993 (the Policy Statement) urges licensees to upgrade plant Technical Specifications by focusing the Technical Specifications on those requirements that are of controlling importance to operational safety. To identify those requirements, the Policy Statement includes four criteria to be used in screening the Technical Specifications. Technical Specifications that satisfy one or more of the criteria must be retained. Specifications that do not satisfy any of the criteria may be removed from the Technical Specifications. The Policy Statement states that removed requirements must be relocated into a licensee-controlled program or procedure. This attachment provides the results of applying the Policy Statement screening criteria to the WCGS Technical Specifications.

Background

The NRC issued an Interim Policy Statement on Technical Specification Improvement, 52 FR 3788, February 6, 1987. In accordance with the Interim Policy Statement, the purpose of Technical Specifications is to impose those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety by establishing those conditions of operation which cannot be changed without prior Commission approval and by identifying those features which are of controlling importance to safety.

The criteria contained in the Interim Policy Statement were applied to the Westinghouse Standard Technical Specifications (STS), NUREG-0452, Revision 4 and Draft Revision 5, and submitted to the NRC in WCAP-11618. The results of the NRC review were issued by letter to the Westinghouse Owners Group dated May 9, 1988.

In July 1993, the NRC issued the Final Policy Statement on Technical Specification Improvements. The Final Policy Statement incorporates the information obtained from public comments and from the experience gained in applying the interim policy criteria during development of new, vendor-specific STS. The new STS for Westinghouse plants are contained in NUREG-1431 issued in September 1992.

Application of the Screening Criteria

Application of the criteria from the Final Policy Statement to the Technical Specifications was begun by preparing a screening form similar to that used in WCAP-11618 except that a separate screening criterion for risk-significant structures, systems, and components was added as required by the Final Policy Statement. Each of the 115 Technical Specifications was evaluated using the screening criteria and the clarifications included in the Policy Statement discussion of each criteria.

During the Technical Specification evaluations, reference was made to the current Westinghouse STS and bases (Ref. 2), the screening forms in WCAP-11618 (Ref. 3), the NRC evaluation of WCAP-11618 (Ref. 4), the results of an NRC test application of screening criteria to the WCGS Technical Specifications (Ref. 5), and the results of applying the interim selection criteria to the North Anna Plant.

Table 1 provides a summary of the results of applying the Final Policy Statement criteria. Table 1 also provides, for comparison, the results of the NRC review of previous Westinghouse STS in Ref. 4. The notes to Table 1 include information regarding the disposition of Technical Specifications and provide supporting information justifying some of the proposed Technical Specification changes.

The screening forms for those Technical Specifications that did not satisfy any of the criteria and, therefore proposed for relocation, are included in Table 2.

Appendix A is the Probabilistic Safety Assessment evaluation that was used to identify structures, systems, and components that satisfied Criterion 4 on the screening forms.

References

In this attachment and on the screening forms, the following references have been used:

1. WCGS Technical Specifications and Bases (NUREG-1136) as amended.

2. Standard Technical Specifications, Westinghouse Plants, NUREG-1431, September 1992.

3. J. D. Andrachek, et. al., Methodically Engineered, Restructured, and Improved Technical Specifications, MERITS Program - Phase II Task 5, Criteria Application, WCAP-11618, November 1987.

4. NRC letter to Westinghouse Owners Group (T. Murley to R. Newton), "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," May 9, 1988.

5. NRC memorandum (V. Stello to NRC Commissioners), "Test Application of TSIP Technical Specification Selection Criteria," February 7, 1986.

6. NRC Generic Letter 85-05, "Inadvertent Boron Dilution Events," January 31, 1985.

7. NSAC-183, "Risk of PWR Reactivity Accident During Shutdown and Refueling."

8. TU Electric Letter to NRC, TXX-93098, dated April 30, 1993, and NRC Approval and SER dated November 3, 1993.

9. TR-92-0063, "Wolf Creek Generating Station Individual Plant Examination Summary Report," September 1992.

10. WASH-1400, "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," October 1975.

TABLE 1					
Summary of Criteria Application Results Reactivity Control Systems					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.1.1.1	3.1.1.1	Shutdown Margin	Retain	See Note 1	1
3.1.1.2	3.1.1.2	Shutdown Margin ≤200 F	Retain	N/A See Note 1	1
3.1.1.3	3.1.1.3	Moderator Temp. Coefficient	Retain	Retain	
3.1.1.4	3.1.1.4	Min. Temperature for Criticality	Retain	Retain	
3.1.2.1	3.1.2.1	Boration Path Shutdown	Relocate	Relocate	
3.1.2.2	3.1.2.2	Boration Path Operating	Relocate	Relocate	
3.1.2.3	3.1.2.3	Charging Pumps Shutdown	Relocate	See Note 2	2
3.1.2.4	3.1.2.4	Charging Pumps Operating	Relocate	Relocate	
3.1.2.5	3.1.2.5	Borated Water Sources Shutdown	Relocate	Relocate	
3.1.2.6	3.1.2.6	Borated Water Sources Operating	Relocate	Relocate	
3.1.3.1	3.1.3.1	Movable Control Assemblies - Group Height	Retain	Retain	1
3.1.3.2	3.1.3.2	Position Indication - Operating	Relocate	Retain	3
3.1.3.3	3.1.3.3	Position Indication - Shutdown	Relocate	Relocate	3
3.1.3.4	3.1.3.4	Rod Drop Time	Relocate	Relocate	4
3.1.3.5	3.1.3.5	Shutdown Rod Insertion Limits	Retain	Retain	
3.1.3.6	3.1.3.6	Control Rod Insertion Limits	Retain	Retain	1

TABLE 1 (Cont.)					
Summary of Criteria Application Results Power Distribution Limits					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.2.1	3.2.1	Axial Flux Differ.	Retain	Retain	
3.2.2	3.2.2	Heat Flux Hot Channel Factor	Retain	Retain	
3.2.3	3.2.3	Nuclear Enthalpy Rise Hot Channel Factor	Retain	Retain	
3.2.4	3.2.4	Quadrant Power Tilt Ratio	Retain	Retain	
3.2.5	3.2.5	DNB Parameters	Retain	Retain	

TABLE 1 (Cont.)					
Summary of Criteria Application Results Instrumentation					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.3.1	3.3.1	Reactor Trip System Instrumentation	Retain	Retain	
3.3.2	3.3.2	Eng. Safety Feature Actuation System Instrumentation	Retain	Retain	
3.3.3.1	3.3.3.1	Radiation Monitoring Instrumentation	Retain	Retain	
3.3.3.2	3.3.3.2	Movable Incore Detectors	Relocate	Relocate	
3.3.3.3	3.3.3.3	Seismic Instrumentation	Relocate	Relocate	
3.3.3.4	3.3.3.4	Meteorological Instrumentation	Relocate	Relocate	
3.3.3.5	3.3.3.5	Remote Shutdown Instrumentation	Retain	Retain	
3.3.3.6	3.3.3.6	Accident Monitoring Instrumentation	Retain	Retain	5
3.3.3.9	3.3.3.9	Loose Parts Detection System	Relocate	Relocate	
3.3.3.11		Explosive Gas Monitoring Instrumentation	Not Reviewed	Relocate	6
3.3.4	3.3.4	Turbine Overspeed Protection	Relocate	Relocate	7

TABLE 1 (Cont.)					
Summary of Criteria Application Results Reactor Coolant System					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.4.1.1	3.4.1.1	Reactor Coolant Loops and Coolant Circulation	Retain	Retain	
3.4.1.2	3.4.1.2	RCS Hot Standby	Retain	Retain	
3.4.1.3	3.4.1.3	RCS Hot Shutdown	Retain	Retain	
3.4.1.4.1	3.4.1.4.1	Cold Shutdown Loops Filled	Retain	Retain	
3.4.1.4.2	3.4.1.4.2	Cold Shutdown Loops Not Filled	Retain	Retain	
3.4.2.1	3.4.2.1	Safety Valves -Shutdown	Relocate	Relocate	
3.4.2.2	3.4.2.2	Safety Valves -Operating	Retain	Retain	
3.4.3	3.4.3	Pressurizer	Retain	Retain	
3.4.4	3.4.4	Relief Valves	Retain	Retain	
3.4.5	3.4.5	Steam Generators	Relocate	Relocate	8
3.4.6.1	3.4.6.1	Leakage Detection Systems	Retain	Retain	
3.4.6.2	3.4.6.2	Operational Leakage	Retain	Retain	
3.4.7	3.4.7	Chemistry	Relocate	Relocate	9
3.4.8	3.4.8	Specific Activity	Retain	Retain	
3.4.9.1	3.4.9.1	Pressure/Temperature Limits	Retain	Retain	
3.4.9.2	3.4.9.2	Pressurizer Pressure/Temperature	Relocate	Relocate	
3.4.9.3	3.4.9.3	Overpressure Protection System	Retain	Retain	
3.4.10	3.4.10	Structural Integrity	Relocate	Relocate	10
3.4.11	3.4.11	RCS Vents	Relocate	Relocate	

TABLE 1 (Cont.)					
Summary of Criteria Application Results Emergency Core Cooling Systems					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.5.1	3.5.1	Accumulators	Retain	Retain	
3.5.2	3.5.2	ECCS Subsystems Tavg ≥ 350°F	Retain	Retain	
3.5.3	3.5.3	ECCS Subsystems Tavg < 350°F	Retain	Retain	
3.5.4		ECCS Subsystems Tavg ≤ 200°F	Not Reviewed	Retain	2, 11
3.5.5	3.5.5	RWST	Retain	Retain	

TABLE 1 (Cont.)					
Summary of Criteria Application Results Containment Systems					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.6.1.1	3.6.1.1	Containment Integrity	Retain	Retain	12
3.6.1.2	3.6.1.2	Containment Leakage	See Note 12	See Note 12	12
3.6.1.3	3.6.1.3	Containment Airlocks	Retain	Retain	
3.6.1.4	3.6.1.5	Internal Pressure	Retain	Retain	
3.6.1.5	3.6.1.6	Air Temperature	Retain	Retain	
3.6.1.6	3.6.1.7	Contain. Vessel Structural Integrity	Relocate	Relocate	13
3.6.1.7	3.6.1.8	Containment Ventilation System	Retain	Retain	14
3.6.2.1	3.6.2.1	Containment Spray System	Retain	Retain	
3.6.2.2	3.6.2.2	Spray Additive System	Retain	Retain	
3.6.2.3		Containment Cooling System	Retain	Retain	
3.6.3	3.6.3	Containment Isolation Valves	Retain	Retain	
3.6.4.1	3.6.4.1	Hydrogen Analyzers	Retain	Delete	15
3.6.4.2	3.6.4.2	Hydrogen Control System	Retain	Retain	

TABLE 1 (Cont.)					
Summary of Criteria Application Results Plant Systems					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.7.1.1	3.7.1.1	Safety Valves	Retain	Retain	
3.7.1.2	3.7.1.2	Auxiliary Feedwater System	Retain	Retain	
3.7.1.3	3.7.1.3	Condensate Storage Tank	Retain	Retain	
3.7.1.4	3.7.1.4	Specific Activity	Retain	Retain	
3.7.1.5	3.7.1.5	Main Steam Isolation Valves	Retain	Retain	
3.7.1.6		Steam Generator Atmospheric Relief Valves	Not Reviewed	Retain	
3.7.1.7		Main Feedwater Isolation Valves	Not Reviewed	Add	16
3.7.2	3.7.2	Steam Generator Pressure/Temperature Limits	Relocate	Relocate	
3.7.3	3.7.3	Component Cooling Water	Retain	Retain	
3.7.4	3.7.4	Essential Service Water System	Retain	Retain	
3.7.5	3.7.5	Ultimate Heat Sink	Retain	Retain	
3.7.6		Control Room Emerg. Ventilation System	Retain	Retain	
3.7.7	3.7.8	Emerg. Exhaust System - Auxiliary Building	Retain	Retain	
3.7.8	3.7.9	Snubbers	Relocate	Relocate	17
3.7.9	3.7.10	Sealed Source Contamination	Relocate	Relocate	
3.7.12	3.7.13	Area Temperature Monitoring	Relocate	Relocate	18

TABLE 1 (Cont.)					
Summary of Criteria Application Results Electrical Power Systems					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.8.1.1	3.8.1.1	AC Sources Operating	Retain	Retain	
3.8.1.2	3.8.1.2	AC Sources Shutdown	Retain	Retain	
3.8.2.1		DC Sources Operating	Retain	Retain	
3.8.2.2		DC Sources Shutdown	Retain	Retain	
3.8.3.1	3.8.3.1	Onsite Power Distrib. - Operating	Retain	Retain	
3.8.3.2	3.8.3.2	Onsite Power Distrib. - Shutdown	Retain	Retain	
3.8.4.1	3.8.4.1	Containment Penetration Conductor Overcurrent Protection Devices	Relocate	Relocate	

TABLE 1 (Cont.)					
Summary of Criteria Application Results Refueling Operations					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.9.1	3.9.1	Boron Concentration	Retain	Retain	
3.9.2	3.9.2	Instrumentation	Retain	Retain	
3.9.3	3.9.3	Decay Time	Retain	Retain	
3.9.4	3.9.4	Containment Building Penetrations	Retain	Retain	
3.9.5	3.9.5	Communications	Relocate	Relocate	
3.9.6	3.9.6	Refueling Machine	Relocate	Relocate	
3.9.7	3.9.7	Crane Travel - Spent Fuel Stor. Facility	Relocate	Relocate	
3.9.8.1	3.9.8.1	RHR and Coolant Recirculation - High Water Level	Retain	Retain	
3.9.8.2	3.9.8.2	RHR and Coolant Recirculation - Low Water Level	Retain	Retain	
3.9.9	3.9.9	Containment Ventilation System	Retain	Retain	
3.9.10.1		Water Level Reactor Vessel - Fuel Assemblies	Retain	Retain	
3.9.10.2		Water Level Reactor Vessel - Control Rods	Not Reviewed	Relocate	19
3.9.11	3.9.11	Water Level -Storage Pool	Retain	Retain	
3.9.12		Spent Fuel Assembly Storage	Not Reviewed	Retain	
3.9.13	3.9.12	Emergency Exhaust System Fuel Building	Retain	Retain	

TABLE 1 (Cont)					
Summary of Criteria Application Results Special Test Exceptions					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.10.1	3.10.1	Shutdown Margin	Relocate	Delete	20
3.10.2	3.10.2	Group Height, Insertion, and Power Distribution Limits	Retain	Retain	
3.10.3	3.10.3	Physics Tests	Retain	Retain	
3.10.4	3.10.4	Reactor Coolant Loops	Retain	Retain	
3.10.5	3.10.5	Position Indication System Shutdown	Relocate	Relocate	20

TABLE 1 (Cont.)					
Summary of Criteria Application Results Radioactive Effluents					
Tech Spec Number	STS Rev. 5 Number	Technical Specification Title	NRC Results	WCGS Results	Note
3.11.1.4	3.11.1.4	Liquid Holdup Tanks	Relocate	Relocate	21
3.11.2.5	3.11.2.5	Explosive Gas Mixture	Relocate	Relocate	6
3.11.2.6	3.11.2.6	Gas Storage Tanks	Relocate	Relocate	21

Notes to Table 1:

NOTES: 1. SDM in Modes 1 and 2 is ensured by the control rods maintained at or above their insertion limits and, for certain events which add positive reactivity, the boration capability of the ECCS is credited. The NRC review issued to the WOG dated May 9, 1988, concluded that the SDM TS could be relocated for Modes 1 and 2 and retained for Modes 3, 4, and 5. However, Wolf Creek has determined that the SDM requirements for Modes 1 and 2 should be retained in the Technical Specifications under other Reactivity Control Systems. The changes to Technical Specification 3.1.1.1 consist of deleting Modes 1 and 2 from the LCO applicability and incorporating the Modes 1 and 2 requirements under new Technical Specification 3.1.1.5 and existing Technical Specifications 3.1.3.1 and 3.1.3.6.

Action a of LCO 3.1.3.6 has been added to address the required actions for a loss of SDM in Modes 1 and 2. New LCO 3.1.3.6 Action a provides one hour to verify SDM or initiate boration, consistent with the timing for Actions 3.1.3.1a and 3.1.3.1c of LCO 3.1.3.1. The Actions for LCO 3.1.1.1 in Modes 3 and 4 and LCO 3.1.1.2 in Mode 5 have been revised to replace "immediately" with "within 15 minutes" to implement boration, per the STS. SR 4.1.1.1.1a for Modes 1 and 2 has been incorporated into Action 3.1.3.1a and 3.1.3.1c of LCO 3.1.3.1. SR 4.1.1.1.1b and Action 3.1.3.1.c.3.b) of LCO 3.1.3.1 have been deleted since they are redundant to renumbered SR 4.1.3.6.1. SR 4.1.1.1.1c, regarding estimated critical position, has been moved to Technical Specification 3.1.3.6, Control Rod Insertion Limits, as SR 4.1.3.6.2. Moving these SDM requirements for Modes 1 and 2 to Technical Specification 3.1.3.1 and 3.1.3.6 improves the specifications by placing actions and surveillances for inoperable rods and insertion limits with their appropriate LCOs.

SR 4.1.1 1.1d and SR 4.1.1.1.2, regarding measuring SDM prior to 5% rated thermal power (RTP) with rods fully inserted and maintaining core reactivity within predicted values, have been converted into new Technical Specification 3.1.1.5, Core Reactivity.

2. SR 4.1.2.3.2 limits the number of operable centrifugal charging pumps to one in Modes 4, 5, and 6 (except when the reactor vessel head is removed). This is an operating restriction of the reactor vessel cold overpressure analysis. This SR will be retained under LCO 3.5.4, ECCS Subsystems - Tavg ≤200°F for Modes 5 and 6. The footnote to 3.1.2.3 is deleted because it is redundant to the footnote for Specification 3.5.4. SR 4.5.3.2 addresses Mode 4.

3. The NRC review of LCO 3.1.3.2 and LCO 3.1.3.3 concluded that they could be relocated. However, if an associated SR is necessary to meet the operability requirements for a retained LCO, the SR should be relocated to the retained LCO. Our evaluation found that LCO 3.1.3.2 is associated with a transient analysis

initial condition and supports LCO 3.1.3.1. As such, LCO 3.1.3.2 will be retained as is. The surveillance associated with LCO 3.1.3.3 is not required for any retained LCO and, therefore, SR 4.1.3.3 will be relocated.

4. The NRC review of this LCO concluded that it could be relocated. However, if an associated SR is necessary to meet the operability requirements for a retained LCO, the SR should be relocated to the retained LCO. SR 4.1.3.4 is required to ensure the operability of control rods under LCO 3.1.3.1 and will be retained under that LCO with the rod drop time limit given in new USAR Section 16.1.3.2. This is consistent with STS.

5. The Regulatory Guide 1.97, Rev. 2, Type A variables identified in USAR Appendix 7A are retained. The neutron flux (Gamma-Metrics) and RVLIS instrumentation will be added. The non-Type A variables are identified and evaluated on the screening form. The relocated instruments are:

Containment Pressure - Extended Range

PZR Safety Valve Position Indication

Unit Vent High Range Noble Gas Monitor

PORV and PORV Block Valve position indicators have been deleted from Technical Specification 3.3.3.6 and monthly channel checks have been added to LCO 3.4.4 as discussed in the Safety Evaluation, Attachment I.

6. This specification will be relocated and an Explosive Gas Monitoring Program statement will be incorporated into new Section 6.8.5.

7. This specification will be relocated and a Turbine Overspeed Protection Reliability Program statement will be incorporated into new Section 6.8.5.

8. This specification will be relocated and a Steam Generator Tube Surveillance Program statement will be included in new Section 6.8.5.

9. This specification will be relocated and a Primary Water Chemistry Program statement will be included in new Section 6.8.5.

10. The LCO will be relocated and the associated SR regarding RCP flywheel integrity will be retained in new Section 6.8.5 as a programmatic requirement.

11. This LCO is intended to prevent a loss of the decay heat removal function in Mode 5 and Mode 6 with the reactor vessel head installed by allowing the safety injection pumps to be operable when the water level is below the vessel flange. The LCO will be retained. Consideration was given to incorporating the

restrictions on pump operation into LCO 3.4.9.3, Overpressure Protection, which would have been in conformance with the STS approach. However, the Modes and RCS temperatures for which these specifications apply prevented combining them into one specification.

12. Containment testing is a requirement imposed by Appendix J of 10 CFR 50. This LCO will be relocated; however, the values of parameters defining leakage limits from 3.6.1.2 will be retained under the Containment Integrity Bases. SR 4.6.1.1c will be modified to eliminate reference to a specification that was relocated, and instead reference corresponding USAR Section 16.6.1.1.

13. This specification will be relocated and a Containment Tendon Surveillance Program statement will be incorporated into new Section 6.8.5.

14. SR 4.6.1.7.2 will be modified to eliminate reference to a specification that was relocated, and instead reference corresponding USAR Section 16.6.1.1.

15. LCO 3.6.4.1 is deleted since it is redundant to LCO 3.3.3.6 and is obsolete per the STS.

16. A new Technical Specification for operability of the Main Feedwater Isolation Valves (MFIVs) will be added to the Technical Specifications for Wolf Creek. The requirements will be identical to those in the Callaway Technical Specifications. Inclusion of a specification for the MFIVs is consistent with NRC Policy Statement Criterion 3 regarding accident mitigating components.

17. This specification will be relocated and a Snubber Inspection Program statement will be included in New Section 6.8.5.

18. This specification will be relocated and an Area Temperature Monitoring Program statement will be included in new Section 6.8.5.

19. This specification places a lower limit on the amount of water above the top of the fuel assemblies in the reactor vessel during movement of control rods. The Bases state that this ensures the water removes 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly in the event of a fuel handling accident during core alterations. However, the movement of control rods is not associated with the initial conditions of a fuel handling accident, and the Bases do not address any concerns regarding inadvertent criticality which could lead to a breach of the fuel rod cladding. Inadvertent criticality during Mode 6 is prevented by maintaining proper boron concentration in the coolant in accordance with LCO 3.9.1. Therefore, this LCO will be relocated.

20. The NRC review concluded that: (1) special test exceptions 3.10.1 through 3.10.4 may be included with corresponding LCOs which are remaining in Technical Specifications, and (2) special test exception 3.10.5 may be relocated along with LCO 3.1.3.3. LCO 3.10.1 is only applicable in Mode 2. As discussed in Note 1 above the SHUTDOWN MARGIN requirements for Modes 1 and 2 are retained in other Reactivity Control System Technical Specifications. Retained Special Test Exceptions 3.10.2 and 3.10.3 address Special Test Exception 3.10.1 for LCOs 3.1.3.1 and 3.1.3.6. Therefore, LCO 3.10.1 will be deleted. Also, per the stated NRC conclusion, LCO 3.10.5 will be relocated. LCOs 3.10.2 through 3.10.4 will be retained as they are.

21. This specification will be relocated and a Storage Tank Radioactivity Monitoring Program statement will be included in Section 6.8.5.

TABLE 2

SCREENING FORMS FOR SPECIFICATIONS TO BE RELOCATED

Screening Forms for the following Technical Specifications are attached:

REACTIVITY CONTROL SYSTEMS

3.1.1.1 SHUTDOWN MARGIN

Shutdown Margin requirements for Modes 1 and 2 will be incorporated under other Reactivity Control System Technical Specifications.

3.1.2.1 FLOW PATHS - SHUTDOWN

3.1.2.2 FLOW PATHS - OPERATING

3.1.2.3 CHARGING PUMPS - SHUTDOWN

3.1.2.4 CHARGING PUMPS - OPERATING

3.1.2.5 BORATED WATER SOURCES - SHUTDOWN

3.1.2.6 BORATED WATER SOURCES - OPERATING

3.1.3.3 POSITION INDICATION SYSTEM - SHUTDOWN

3.1.3.4 ROD DROP TIME

POWER DISTRIBUTION LIMITS

NONE

INSTRUMENTATION

3.3.3.2 MOVABLE INCORE DETECTORS

3.3.3.3 SEISMIC INSTRUMENTATION

3.3.3.4 METEOROLOGICAL INSTRUMENTATION

3.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

3.3.3.9 LOOSE-PART MONITORING INSTRUMENTATION

3.3.3.11 EXPLOSIVE GAS MONITORING INSTRUMENTATION

3.3.4 TURBINE OVERSPEED PROTECTION

REACTOR COOLANT SYSTEM

3.4.2.1	SAFETY VALVES - SHUTDOWN
3.4.5	STEAM GENERATORS
3.4.7	CHEMISTRY
3.4.9.2	PRESSURIZER P/T LIMITS
3.4.10	STRUCTURAL INTEGRITY
3.4.11	REACTOR COOLANT SYSTEM VENTS

EMERGENCY CORE COOLING SYSTEMS

NONE

CONTAINMENT SYSTEMS

3.6.1.2	CONTAINMENT LEAKAGE
3.6.1.6	CONTAINMENT VESSEL STRUCTURAL INTEGRITY

PLANT SYSTEMS

3.7.2	STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION
3.7.8	SNUBBERS
3.7.9	SEALED SOURCE CONTAMINATION
3.7.12	AREA TEMPERATURE MONITORING

ELECTRICAL POWER SYSTEMS

3.8.4.1	CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

REFUELING OPERATIONS

3.9.5	COMMUNICATIONS
3.9.6	REFUELING MACHINE
3.9.7	CRANE TRAVEL - SPENT FUEL STORAGE FACILITY
3.9.10.2	WATER LEVEL - REACTOR VESSEL/CONTROL RODS

SPECIAL TEST EXCEPTIONS

3.10.1 SHUTDOWN MARGIN

3.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

RADIOACTIVE EFFLUENTS

3.11.1.4 LIQUID HOLDUP TANKS

3.11.2.5 EXPLOSIVE GAS MIXTURE

3.11.2.6 GAS STORAGE TANKS

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.1.1.1 SHUTDOWN MARGIN
Applicable Modes: 1, 2, 3, and 4

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
*	*	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
	*		Based on the discussion below, this LCO satisfies criterion 2 for Modes 3, 4, and 5. For Modes 1 and 2, the criterion is not satisfied.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

Shutdown Margin (SDM) requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCOs 3.1.3.5 and 3.1.3.6, for rod insertion. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

The Bases for this TS state that sufficient SDM ensures (1) the reactor can be made subcritical from all operating conditions, (2) reactivity transients associated with postulated accidents are controllable within acceptable limits , and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. The most restrictive condition is EOL at no load operating Tavg associated with a MSLB. A minimum SDM of 1.3% Delta-k/k is required to control the reactivity added by the cooldown. The SDM requirements must also protect against:

a. Inadvertent boron dilution,
b. An uncontrolled rod withdrawal from subcritical or low power condition,
c. Startup of an inactive reactor coolant pump, and
d. Rod ejection.

In Modes 1 and 2, SDM is verified by observing that the requirements for rod insertion limits are met. In Modes 3, 4, and 5, SDM is verified by performing a reactivity balance calculation.

The SDM (boration control) TS is not applicable to a process variable indicating in the control room a significant degradation of the RCPB. Therefore, SDM does not satisfy criterion 1.

SDM is an initial condition of accident and transient analyses. However, during operation in Modes 1 and 2, the available SDM is determined by the rod insertion limits. Therefore, SDM (boration control) requirements are not applicable to a process variable, design feature, or operating restriction that either assumes the failure of or presents a challenge to the integrity of a fission product barrier and, thus, do not satisfy criterion 2 for these operating Modes. However, this TS does satisfy criterion 2 for Modes 3 and 4.

The TS requirements for SDM are not applicable to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient; these requirements, therefore, do not satisfy criterion 3.

Ref. 4 concluded that the LCO could be relocated for Modes 1 and 2 but must be retained for Modes 3, 4, and 5. However, Wolf Creek has determined that the SDM requirements for Modes 1 and 2 should be retained in the Technical Specifications under other Reactivity Control Systems. The changes to Technical Specification 3.1.1.1 consist of deleting Modes 1 and 2 from the LCO applicability and incorporating the Modes 1 and 2 requirements under new Technical Specification 3.1.1.5 and existing Technical Specifications 3.1.3.1 and 3.1.3.6.

Action a of LCO 3.1.3.6 has been added to address the required actions for a loss of SDM in Modes 1 and 2. New LCO 3.1.3.6 Action a provides one hour to verify SDM or initiate boration, consistent with the timing for Actions 3.1.3.1a and 3.1.3.1c of LCO 3.1.3.1. The Actions for LCO 3.1.1.1 in Modes 3 and 4 and LCO 3.1.1.2 in Mode 5 have been revised to replace "immediately" with "within 15 minutes" to implement boration per the STS. SR 4.1.1.1.1a for Modes 1 and 2 has been incorporated into Actions 3.1.3.1a and 3.1.3.1c of LCO 3.1.3.1. SR 4.1.1.1.1b and Action 3.1.3.1.c.3.b) of LCO 3.1.3.1 have been deleted since they are redundant to renumbered SR 4.1.3.6.1. SR 4.1.1.1.1c, regarding estimated critical position, has been moved to Technical Specification 3.1.3.6, Control Rod Insertion Limits, as SR 4.1.3.6.2. Moving these SDM requirements for Modes 1 and 2 to Technical Specifications 3.1.3.1 and 3.1.3.6 improves the specifications by placing actions and requirements for inoperable rods and insertion limits with their appropriate LCOs.

SR 4.1.1.1.1d and SR 4.1.1.1.2, regarding measuring SDM prior to 5% RTP with rods fully inserted and maintaining core reactivity within predicted values, have been converted into new Technical Specification 3.1.1.5, Core Reactivity.

From Reference 2, shutdown margins during power operation have not been shown to be risk significant to public health and safety by either operating experience or PSA. Boron dilution accidents, rod ejection accidents and return to power from plant transients are the major scenarios for which the shutdown margin is needed. However, none of the three are dominant risk contributors. Shutdown margins are not modeled in the Wolf Creek IPE. Therefore, this TS does not satisfy Criterion 4.

Based on the above, the SDM requirements for Modes 1 and 2 will be retained under other Reactivity Control System Technical Specifications. The LCO for Modes 3, 4, and 5 should be retained because SDM in these modes is not verified by the rod insertion limits.

(4) **CONCLUSION**

X This Technical Specification is retained.

_ The Technical Specification may be relocated to the following controlled document(s):

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.1.2.1 BORATION FLOW PATHS - SHUTDOWN
Applicable Modes: 4, 5, and 6

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The Bases for this LCO state that the purpose is to assure negative reactivity control is available during each Mode of facility operation.

The purpose of the boration subsystem of the CVCS is to provide the means to control the boron concentration to maintain SDM. The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of an assumed boron dilution event, the automatic response of the BDMS, or that required of the operator, is to close the appropriate valves in the reactor makeup system before the shutdown margin is lost. Automatic actuation of the boration subsystem is not required to mitigate the event. Shutdown Margin (SDM) requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCOs 3.1.3.5 and 3.1.3.6, for rod insertion. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

Ref. 5 notes that the normal capability to control reactivity with boron is not credited in the accident analysis.

The boration subsystem TS is not applicable to installed instrumentation used to detect or indicate a significant degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not associated with a process variable that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. As stated in the analyses of boron dilution events, the BDMS performs automatic actions in response to detecting an assumed boron dilution event. These actions are credited for events occurring in Modes 3, 4, and 5. The actions include providing an alarm, automatically isolating the dilution flow path, and automatically initiating boration of the RCS from the RWST via the charging pumps. For these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. This is desirable, but beyond the scope of a primary success path action. Therefore, this TS does not satisfy criterion 2.

The boration subsystem TS does not apply to any SSC that is a part of the primary success path and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; therefore, this TS does not satisfy criterion 3. Ref. 3 also notes that operability of the charging pumps, the RWST, and associated flowpaths is required as part of the ECCS TS.

The shutdown flow paths, used to inject borated water to maintain SDM, have not been shown to be significant to public health and safety by either operating experience or PSA. The shutdown flow paths are modeled in the WCGS IPE for the ATWS event, in the LTS fault tree. However, the core damage values for the ATWS event sequences are extremely low, well below the NRC screening value of 1.0E-06. Therefore, this TS does not satisfy Criterion 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.1.2.2 BORATION FLOW PATHS - OPERATING
Applicable Modes: 1, 2, and 3

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The Bases for this LCO state that the purpose is to assure negative reactivity control is available during each Mode of facility operation.

The purpose of the boration subsystem of the CVCS is to provide the means to control the boron concentration to maintain SDM. The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of an assumed boron dilution event in Mode 3, the automatic response of the BDMS, or that required of the operator, is to close the appropriate valves in the reactor makeup system before the SDM is lost. Automatic actuation of the boration subsystem is not required to mitigate the event. In Modes 1 and 2, a dilution event is initially mitigated by the RTS and the reactor is shut down by insertion of the control rods. Continued dilution will tend to take the reactor critical; however, the operator has more than 30 minutes to stop the dilution flow. Ref. 5 notes that the normal capability to control reactivity with boron is not credited in the accident analysis. Shutdown Margin (SDM) requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCOs 3.1.3.5 and 3.1.3.6, for rod insertion. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

Based on the foregoing, the boration subsystem TS is not applicable to installed instrumentation used to detect or indicate a significant degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not associated with a process variable that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. As stated in the analyses of boron dilution events, the BDMS performs automatic actions in response to detecting an assumed boron dilution event. These actions are credited for events occurring in Mode 3. The actions include providing an alarm, automatically isolating the dilution flow path, and automatically initiating boration of the RCS from the RWST via the charging pumps. For these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. This is desirable, but beyond the scope of a primary success path action. In Modes 1 and 2, the operator is required to isolate the dilution flow path subsequent to a reactor trip. Therefore, the boration subsystem is not a design feature required to be operable to mitigate these events, and this TS does not satisfy criterion 2.

The boration subsystem TS does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; therefore, the TS does not satisfy criterion 3. Ref. 3 also notes that operability of the charging pumps, the RWST, and associated flowpaths is required as part of the ECCS TS.

For the MSLB event, the sequence of events takes the plant to cold shutdown conditions and, therefore, boration of the RCS is necessary. However, the boration flowpath in this case is required as part of the ECCS function.

From Reference 2, the operating flow paths, used to inject borated water to maintain SDM, have not been shown to be significant to public health and safety by either operating experience or PSA. While the operating flow paths are not modeled in the Wolf Creek IPE, they are similar to the shutdown flow paths, which have been modeled in the Wolf Creek IPE and which have extremely low risk values. Therefore, this TS does not satisfy Criterion 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.1.2.3 CHARGING PUMPS - SHUTDOWN
Applicable Modes: 4, 5, and 6

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The Bases for this LCO state that the purpose is to assure negative reactivity control is available during each Mode of facility operation. Equipment required to perform this function includes: (1) borated water sources, (2) CCPs, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power source from the EDGs.

The purpose of the boration subsystem of the CVCS is to provide the means to control the boron concentration to maintain SDM. The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of an assumed boron dilution event, the automatic response of the BDMS, or that required of the operator, is to close the appropriate valves in the reactor makeup system before the shutdown margin is lost. Automatic actuation of the boration subsystem is not assumed to mitigate the event. Ref. 5 notes that the normal capability to control reactivity with boron is not credited in the accident analysis. Shutdown Margin (SDM) requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCOs 3.1.3.5 and 3.1.3.6, for rod insertion. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

The boration subsystem TS is not applicable to installed instrumentation used to detect or indicate a significant degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not associated with a process variable that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. As stated in the analyses of boron dilution events, the BDMS performs automatic actions in response to detecting an assumed boron dilution event. These actions are credited for events occurring in Modes 3, 4, and 5. The actions include providing an alarm, automatically isolating the dilution flow path, and automatically initiating boration of the RCS from the RWST via the charging pumps. As stated in Ref. 3 for these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. This is desirable, but beyond the scope of a primary success path action. Therefore, this TS does not satisfy criterion 2.

The boration subsystem TS does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; therefore, the TS does not satisfy criterion 3. Ref. 3 also notes that operability of the charging pumps, the RWST, and associated flowpaths is required as part of the ECCS TS.

This TS also serves to prevent a cold overpressure event from occurring by limiting the number of operable CCPs to one in Modes 4, 5, and 6 except when the reactor vessel head is removed. This restriction is part of an initial condition for the cold overpressure analysis. The specific SR that imposes this restriction will be incorporated into TS 3/4.5.4 for Modes 5 and 6. TS 4.5.3.2 addresses Mode 4.

From Reference 2 the charging pumps, used to inject borated water into the RCS to maintain SDM, have not been shown to be significant to public health and safety by either operating experience or PSA. The charging pumps have been modeled in the LTS fault tree which is a top event for the ATWS initiating event. The core damage values for the ATWS event are extremely low, well below the NRC screening value. In addition, WCGS IPE comparison with NUMARC 93-01 Section 9.3.1, using both the Risk Reduction and Risk Achievement methods, demonstrated that the CCPs are not risk significant. Therefore, this TS does not satisfy Criterion 4.

(4) CONCLUSION

X This Technical Specification is retained.

SR 4.1.2.3.2 will be retained and incorporated into TS 3/4.5.4.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.1.2.4 CHARGING PUMPS - OPERATING
Applicable Modes: 1, 2, and 3

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The Bases for this LCO state that the purpose is to assure negative reactivity control is available during each Mode of facility operation. The equipment required to perform this function includes: (1) borated water sources, (2) CCPs, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from operable EDGs.

Ref. 3 states that the purpose of the boration subsystem of the CVCS is to provide the means to control the boron concentration to maintain SDM. The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of an assumed boron dilution event in Mode 3, the automatic response of the BDMS, or that required of the operator, is to close the appropriate valves in the reactor makeup system before the SDM is lost. Automatic actuation of the boration subsystem is not assumed to mitigate the event. In Modes 1 and 2, a dilution event is initially mitigated by the RTS and the reactor is shut down by insertion of the control rods. Continued dilution will tend to take the reactor critical; however, the operator has more than 30 minutes to stop the dilution flow. Ref. 5 notes that the normal capability to control reactivity with boron is not credited in the accident analysis. Shutdown Margin (SDM) requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCOs 3.1.3.5 and 3.1.3.6, for rod insertion. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

Based on the foregoing, the boration subsystem TS is not associated with installed instrumentation used to detect or indicate a significant degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. As stated in the analyses of boron dilution events, the BDMS performs automatic actions in response to detecting an assumed boron dilution event. These actions are credited for events occurring in Mode 3. The actions include providing an alarm, automatically isolating the dilution flow path, and automatically initiating boration of the RCS from the RWST via the charging pumps. For these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. Although desirable, this is beyond the scope of a primary success path action. In Modes 1 and 2, the operator is required to isolate the dilution flow path subsequent to a reactor trip. Therefore, the boration subsystem is not required to be operable to mitigate these events, and the TS does not satisfy criterion 2.

The boration subsystem is not a system that is part of the primary success path and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; therefore, the TS does not satisfy criterion 3. Ref. 3 also notes that operability of the charging pumps, the RWST, and associated flowpaths is required as part of the ECCS TS.

For the MSLB event, the sequence of events takes the plant to cold shutdown conditions and, therefore, boration of the RCS is necessary. However, the boration flowpath in this case is required as part of the ECCS function.

From Reference 2, the charging pumps, used to inject borated water into the RCS to maintain SDM, have not been shown to be significant to public health and safety by either operating experience or PSA. While the charging pumps have not been modeled for this application in the WCGS IPE, the application is similar to the shutdown mode application which is included in the ATWS model. The core damage values for the ATWS event are extremely low, well below the NRC cutoff values. In addition, the WCGS IPE comparison with NUMARC 93-01 Section 9.3.1, using both the Risk Reduction and Risk Achievement methods, demonstrated that the CCPs are not risk significant. Therefore, this TS does not satisfy Criterion 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.1.2.5 BORATED WATER SOURCE - SHUTDOWN
Applicable Modes: 5 and 6

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The Bases for this LCO state that the purpose is to assure negative reactivity control is available during each Mode of facility operation. Equipment required to perform this function includes, depending on operating conditions, a combination of: (1) borated water sources, (2) CCPs, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power source from the EDGs.

The purpose of the boration subsystem of the CVCS is to provide the means to control the boron concentration to maintain SDM. The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of an assumed boron dilution event, the automatic response of the BDMS, or that required of the operator, is to close the appropriate valves in the reactor makeup system before the shutdown margin is lost. Automatic actuation of the boration subsystem is not assumed to mitigate the event. Ref. 5 notes that the normal capability to control reactivity with boron is not credited in the accident analysis.

Shutdown Margin (SDM) requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCOs 3.1.3.5 and 3.1.3.6, for rod insertion. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

The boration subsystem TS is not applicable to installed instrumentation used to detect or indicate a significant degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not applicable to a process variable, design feature, or operating restriction that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The boration subsystem TS does not apply to an SSC that is part of the primary success path and functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As stated in the analyses of boron dilution events, the BDMS performs automatic actions in response to detecting an assumed boron dilution event. These actions are credited for events occurring in Modes 3, 4, and 5. The actions include providing an alarm, automatically isolating the dilution flow path, and automatically initiating boration of RCS from the RWST via the charging pumps. As stated in Ref. 3 for these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. Although desirable, this is beyond the scope of a primary success path action. Therefore, this LCO does not satisfy criterion 3.

Ref. 3 also notes that operability of the charging pumps, the RWST, and associated flowpaths is required as part of the ECCS TS.

From Reference 2, the borated water sources have not been shown to be significant to public health and safety by either operational experience or PSA. LER 482/90-025-00 was written to report that a common return line for the Safety Injection pumps to the refueling water storage tank (RWST) had frozen. This could have caused the loss of the RWST. However, design changes have been implemented which will preclude this from occurring in the future. The boric acid tank (BAT), due to it's generally rugged, simple design, has a low failure probability. It was not included in the WCGS IPE, which in general, does not model shutdown. The RWST is not included in the WCGS IPE model for the boron dilution accident during shutdown, but is modeled for accident mitigation while operating. Due to it's simple design and passive function, the RWST has a low failure probability, well below the NRC screening criterion. Therefore, this TS does not satisfy Criterion 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.1.2.6 BORATED WATER SOURCES - OPERATING
Applicable Modes: 1, 2, 3, and 4

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The Bases for this LCO state that the purpose is to assure negative reactivity control is available during each Mode of facility operation. The equipment required to perform this function includes, depending upon operating conditions, combinations of: (1) borated water sources, (2) CCPs, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from operable EDGs.

The purpose of the boration subsystem of the CVCS is to provide the means to control the boron concentration to maintain SDM. The boration subsystem is not assumed to operate to mitigate the consequences of a DBA or transient. In the case of an assumed boron dilution event in Mode 3 or 4, the automatic response of the BDMS, or that required of the operator, is to close the appropriate valves in the reactor makeup system before the SDM is lost. Automatic actuation of the boration subsystem is not assumed to mitigate the event. In Modes 1 and 2, a dilution event is initially mitigated by the RTS and the reactor is shut down by insertion of the control rods. Continued dilution will tend to take the reactor critical; however, the operator has more than 30 minutes to stop the dilution flow and maintain SDM. Shutdown Margin (SDM) requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences. The SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and control rods, assuming that the single rod assembly of highest worth is fully withdrawn. During power operation, SDM control is ensured by operating with the shutdown banks fully withdrawn and the control banks within the limits of LCOs 3.1.3.5 and 3.1.3.6, for rod insertion. When the unit is in the shutdown and refueling modes, the SDM requirements are met by means of adjustments to the RCS boron concentration.

Ref. 5 notes that the normal capability to control reactivity with boron is not credited in the accident analysis.

Based on the foregoing, the boration subsystem TS is not applicable to installed instrumentation used to detect or indicate a significant degradation of the RCPB, therefore, this TS does not satisfy criterion 1.

The boration subsystem TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of an event that assumes failure of or challenges the integrity of a fission product barrier. Thus, the TS does not satisfy criterion 2.

The boration subsystem TS does not apply to a system that is part of the primary success path and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. As stated in the analyses of boron dilution events, the BDMS performs automatic actions in response to detecting an assumed boron dilution event. These actions are credited for events occurring in Modes 3 and 4. The actions include providing an alarm, automatically isolating the dilution flow path, and automatically initiating boration of RCS from the RWST via the charging pumps. For these events, the primary success path for mitigation includes isolating the dilution flowpath. The subsequent actuation of equipment to establish a boron injection flowpath is intended to regain the required SDM. Although desirable, this is beyond the scope of a primary success path action. In Modes 1 and 2, the operator is required to isolate the dilution flow path subsequent to a reactor trip. Therefore, the boration subsystem is not required to be operable to mitigate these events, and the TS does not satisfy criterion 3.

Ref. 3 also notes that operability of the charging pumps, the RWST, and associated flowpaths is required as part of the ECCS TS.

For the MSLB event, the sequence of events takes the plant to cold shutdown conditions and, therefore, boration of the RCS is necessary. However, the boration flowpath in this case is required as part of the ECCS function.

From Reference 2, the borated water sources have not been shown to be significant to public health and safety by either operational experience or PSA. LER 482/90-025-00 was written to report that a common return line from the Safety Injection pumps to the refueling water storage tank (RWST) had frozen. This could have caused the loss of the RWST. However, design changes have been implemented which will preclude this from occurring in the future. The boric acid tank (BAT), due to it's generally rugged, simple design, was not modeled in the WCGS IPE. The RWST is included in the WCGS IPE model but, due to it's simple design and passive function, has a low failure probability, well below NRC screening criteria. Therefore, this TS does not satisfy Criterion 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.1.3.3 POSITION INDICATING SYSTEMS - SHUTDOWN
Applicable Modes: 3, 4, and 5 with the Reactor Trip Breakers Closed

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

Control rod position is used by the operator to verify that the rods are correctly positioned and to verify that the rods are inserted into the core following a reactor trip. Rod position is also used during a reactor startup.

Operability of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the rod alignment and insertion limits. These rod alignment requirements are applicable during power operation to maintain power distribution limits. Rod insertion limits are required to maintain SDM during Modes 1 and 2. The Bases do not address the shutdown condition. The LCO requires that one position indicator be operable to determine the position of any rod not fully inserted. Rod position indication may be used during a control rod withdrawal event from shutdown condition, but it is not required to be operable as an initial condition of a mitigating signal.

The position indication system TS is not applicable to installed instrumentation used to detect and indicate in the control room significant abnormal degradation of the RCPB. Therefore, this TS does not satisfy criterion 1.

The position indication system TS, for shutdown conditions, is not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, the TS does not satisfy criterion 2.

Finally, the position indication system TS does not apply to an SSC that is part of the primary success path and which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, the TS does not satisfy criterion 3.

From Reference 2, the control rod position indicating systems have not been shown to be significant to public health and safety by either operational experience or PSA. The Zion PRA study (Reference 5) included this system, and it was shown to not be risk significant for their plant. The system is not modeled in the WCGS IPE, but there is no indication that it would be identified as risk significant if it were included in the WCGS IPE model. Therefore, this TS does not satisfy Criterion 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

3135.doc

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.1.3.4 ROD DROP TIME
Applicable Modes: 1 and 2

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component th t is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The Bases state that this TS ensures the control rod drop times are consistent with the assumptions of the safety analyses. Therefore, the drop time may be considered a variable that is an initial condition of several events that could present a challenge to a fission product barrier. However, this parameter cannot be monitored, controlled, or maintained within the bounds of the safety analysis by the plant operators. Also, this parameter is one that contributes to the definition of an operable control rod; however, rod drop time is not used to define an operable control rod during plant operation in Modes 1 and 2.

Ref. 3 determined that this specification is not installed instrumentation used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB. Nor is it an SSC that is part of the primary success path and which functions to mitigate any event. Ref. 3 also stated that Rod Drop Time is a variable that is an initial condition of a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Ref. 4 concluded that this LCO may be relocated but the associated SR should be relocated to a retained LCO if the SR is necessary to meet the operability requirements of an LCO. Ref. 2 has relocated this LCO but included the rod drop time limit and conditions required for measuring it as an SR under an LCO for rod alignment.

The rod drop time TS is not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB. Therefore, this TS does not satisfy criterion 1.

The rod drop time TS is associated with a design feature (rod insertion time) that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, rod drop time is not a parameter that is maintained, during plant operations, within the bounds assumed in the accident analyses. Therefore, this TS does not satisfy criterion 2.

The rod drop time TS does apply to an SSC (operable control rod) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, this parameter is not used to define an operable control rod during plant operation. Therefore, this TS does not satisfy criterion 3.

From Reference 2, the rod drop time TS has not been shown to be significant to public health and safety by either operational experience or PSA. While not modeled directly in the WCGS IPE, the ATWS event which is a part of the WCGS IPE model, would provide a conservative approximation of the excessive rod drop time scenario. The ATWS event has a worst case coremelt sequence frequency well below the NRC cutoff value. Therefore, this TS does not satisfy Criterion 4.

(4) **CONCLUSION**

X This Technical Specification is retained.

Rod drop time and plant conditions for measurement will be relocated as an SR under LCO 3.1.3.1.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.3.3.2 MOVABLE INCORE DETECTORS
Applicable Modes: Refer to new USAR Section 16.3.1.1

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

This LCO requires the movable incore detectors to be operable, within defined conditions, whenever the system is used for recalibration of excore detectors, monitoring the quadrant power tilt ratio, or measurement of F_Q and F-Delta H. If the system is not operable, the required action is not to use the system for these purposes. The requirements for maintaining F_Q and F-Delta H within limits are addressed in the TS for power distribution limits.

Ref. 1 states that the operability of the movable incore detectors ensures the accurate measurement of spatial neutron flux distribution of the core.

Ref. 3 notes that the movable incore detector system is not installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB. Also, the system is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Further, the movable incore detector system is not an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The movable incore detector TS is not applicable to installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the RCPB.

The movable incore detector TS is associated indirectly with an operating restriction (flux distribution limits) that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, this operating restriction is maintained by other TS requirements.

The movable incore detector TS does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

From Reference 2, the movable incore detectors have not been shown to be significant to public health and safety by either operational experience or PSA. The detectors are used only for periodic surveillance of the core power distribution and for calibration of the excore detectors and do not initiate any automatic protection action. The detectors are not modeled in the WCGS IPE.

Based on the above, the LCO does not satisfy criteria 1, 2, 3 or 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.3.3.3 SEISMIC INSTRUMENTATION
Applicable Modes: At all times

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The TS Bases state that the seismic monitoring instruments are to determine the magnitude of a seismic event so that the measured response of the plant can be compared to the response used in the design basis and determine if a shutdown is required in accordance with 10 CFR 100. The occurrence of a seismic event would represent a challenge to fission product barriers. However, the ability of the plant to withstand an SSE is a design requirement. The seismic monitoring instrumentation performs no role in mitigating a seismic event or in achieving a safe shutdown condition after a seismic event has occurred.

Ref. 3 determined that the seismic instrumentation is not installed instrumentation that is used to detect degradation of the RCPB. Seismic instrumentation is not assumed to function in the safety analysis and is not an SSC that is part of the primary success path and which function or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The seismic instrumentation TS is not applicable to installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the RCPB.

The seismic instrumentation is not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The seismic instrumentation TS does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

From Reference 2, the seismic instrumentation has not been shown to be significant to public health and safety by either operational experience or PSA. The seismic instrumentation is not designed to monitor seismic events that are of sufficient severity to be a dominant plant risk. The seismic instrumentation is not modeled in the WCGS IPE.

Based on the above, the seismic instrumentation requirements do not satisfy criteria 1, 2, 3, or 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.3.3.4 METEOROLOGICAL INSTRUMENTATION
Applicable Modes: At all times

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The meteorological instrumentation ensures that data is available to estimate potential radiological doses to the public from accidental or routine releases of radioactive materials to the atmosphere. The instrumentation is used to assess the need for recommending protective measures following an accident. The meteorological instrumentation is not used to mitigate a DBA or transient.

Ref. 3 evaluated this instrumentation and concluded that it is not installed instrumentation that is used to detect degradation of the RCPB. Neither is it assumed to function in the safety analysis and is not an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The meteorological instrumentation TS is not applicable to installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the RCPB. Therefore, this TS does not satisfy criterion 1.

The meteorological instrumentation TS is also not associated with a process variable, design feature or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 2.

The meteorological instrumentation TS does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

From Reference 2, the meteorological instrumentation has not been shown to be significant to public health and safety by either operational experience or PSA. Offsite dose calculations for large accidental releases of radioactive materials rely on conservative meteorological and evacuation assumptions and do not take credit for the meteorological instruments cited in this TS to guide emergency measures to protect the public. Finally, per Reference 1, no severe radioactive releases per the PWR Release Category 4, as defined in WASH-1400 (Reference 6) were found to exist at WCGS.

Based on the above, the meteorological instrumentation does not satisfy criteria 1, 2, 3, or 4. This is consistent with the NRC's conclusion in Ref. 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.3.3.6 ACCIDENT MONITORING INSTRUMENTATION
Applicable Modes: 1, 2, and 3

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
*	*	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
*	*	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

* The instrumentation that satisfies criteria 3 or 4 are the Type A variables in USAR Appendix 7A, as well as the risk-significant variables listed in the discussion below. Some of the TS 3/4.3.3.6 instruments may be relocated to USAR Chapter 16. Others must be retained in the Technical Specifications. The Neutron Flux Monitors and the Reactor Vessel Water Level Indicating System (RVLIS) will be added to the Technical Specifications.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

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. The instrumentation allows the operator to verify the response of automatic safety systems and to take preplanned manual actions to accomplish a safe plant shutdown.

The accident monitoring instrumentation is not intended to be a leading indicator of RCS leakage. Although accident monitoring instruments respond to the consequences of a LOCA, the instruments captured by criterion 1 are those that are intended to prevent a LOCA from occurring and to give some indication of RCS leakage prior to the LOCA. Therefore, accident monitoring instrumentation TS is not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB and does not satisfy criterion 1.

Accident monitoring instrumentation is not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Although some variables that are accident monitoring instruments may also establish initial conditions at the time of a DBA or transient (for example, pressurizer level), the post-event function is separate and distinct from the pre-event function. Therefore, the accident monitoring instrumentation TS does not satisfy criterion 2.

Specific accident monitoring instrumentation provides the operator with information needed to perform the required manual actions to bring the plant to a stable condition following an accident. This instrumentation is a component of the primary success path and functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, specific accident monitoring instrumentation satisfies criterion 3.

Ref. 4 states that accident monitoring instrumentation that satisfies the definition of Type A variables in Regulatory Guide 1.97 satisfies criterion 3 and should be retained in the TS. Ref. 4 also states that non-Type A, Category 1 instruments are to be evaluated for inclusion in the TS based on results of risk analyses. In accordance with USAR Table 7A-2, the instruments that are either (1) Type A or (2) non-Type A, Category 1 are:

INSTRUMENT	TS	TYPE A	CATEGORY
Neutron Flux			1
Core Exit Temperature	Yes		1
Reactor Vessel Level			1
RCS T-Cold	Yes	Yes	1
RCS T-Hot	Yes	Yes	1
RCS Pressure	Yes	Yes	1
Pressurizer Level	Yes	Yes	1
RWST Level	Yes	Yes	2
Steam Generator Level-Wide Range	Yes		1
Steam Generator Level-Narrow Range	Yes	Yes	1
Steam Line Pressure	Yes	Yes	1
Condensate Storage Tank Level (Pressure)			1
Containment Pressure	Yes	Yes	1
Containment Pressure- Extended Range	Yes		1
Containment Normal Sump Level	Yes	Yes	1
Containment RHR Sump Level			1
Containment Isolation Valve Position			1
Containment Hydrogen Concentration	Yes		1
Containment Area Radiation	Yes	Yes	1
Radiation Level in RCS (Sampling System)			1
Auxiliary Feedwater Flow Rate	Yes		2
PORV Position Indicator	Yes		2
PORV Block Valve Position Indicator	Yes		N.A.
Safety Valve Position Indicator	Yes		2
Unit Vent-High Range Noble Gas Monitor	Yes		2

The Type A variables satisfy criterion 3 and will not be relocated. The Non-Type A variables are evaluated in the following paragraphs.

1. **Neutron Flux-Source Range**

   Neutron flux is a R.G. 1.97 Category 1, Type B variable. In the emergency operating procedures (EOPs), neutron flux is the specified means to verify reactor subcriticality and is to be monitored during EOP usage. Indication of significant post-trip power generation results in entry into a Function Restoration Procedure (FRP) designed to ensure adequate shutdown reactivity. Based on the significance of this variable in the EOPs, neutron flux will be incorporated into the TS.

2. **Core Exit Temperature**

   This indication is important for determining inadequate core cooling and will be retained.

3. **Reactor Vessel Level**

The EOPs make use of a number of reactor vessel level indicating system (RVLIS) setpoints related to RCS inventory control and indication of inadequate core cooling. These include

a. Indication of inadequate core cooling
b. An alternate to RCS subcooling and pressurizer level as a safety injection initiation criterion.
c. A means of controlling charging flow if pressurizer level indication is not available.
d. A means of determining if RHR operation will be effective based on collapsed liquid level.

The detection of inadequate core cooling represents a potential near term breach of the fuel cladding integrity. Use of reactor vessel level indication in the EOPs includes events that are DBAs. Based on this information, reactor vessel level will be incorporated into the TS.

4. **Steam Generator Level-Wide Range**

Steam generator level-wide range is used in the EOPs as an indicator of steam generator (SG) dryout and as a criterion for establishing feed and bleed cooling of the RCS. Loss of SG level does not, in and of itself, represent an approach to a breach of a fission product barrier. This instrumentation does provide information required to perform a manual action which preserves a critical safety function (heat sink). The steam generator wide range level monitors are modeled in the operator actions OPA-OFB and OPA-OFC, operator feed and bleed action, which are used in the small LOCA, SGTR, secondary break and the TRA, TRO and LSP event trees. The most probable core damage sequence containing this operator action has a frequency of 1.4E-09/year, which is below the NRC significance criteria. However, due to the above discussion, these indicators will be retained

5. **Condensate Storage Tank Level (Pressure)**

This variable is R.G. 1.97 Category 1 if needed to ensure water supply for the AFW system. However it may be Category 3 if the CST is not the primary source of supply. The primary source of AFW supply is the ESW system, and the AFW pump suction lines are automatically transferred to the ESW system upon loss of CST level as indicated by low pressure in the pump suction lines. Since there is no manual action required for switchover to the alternate source of auxiliary feedwater (ESW system), the CST level measurement is not a Type A variable. The condensate storage tank level indicator has not been shown to be significant to public health and safety by either operational experience or PSA. It is not included in the WCGS IPE model. Therefore, the CST level (suction pressure) need not be added to the TS.

6. **Containment Pressure-Extended Range**

R.G. 1.97 defines the purpose of this variable as "detection of potential for or actual breach; accomplishment of mitigation". The EOPs do not base any decisions or actions on this variable. All actions related to containment pressure are based on the normal range containment pressure indication which is a Type A variable. Extended range pressure is not required to take appropriate actions to ensure the integrity of any fission product barrier. The extended range containment pressure indicators have not been shown to be significant to public health and safety by either operational experience or PSA. They are not included in the WCGS IPE model. This TS does not satisfy criterion 3 or 4 and will be relocated.

7. **Containment RHR Sump Level**

This parameter is not a Type A variable. It is a Type B, Category 1 variable. The Containment Normal Sump Level is a Type A, Category 1 variable that will remain in the TS. Although the RHR sump level could be used for event identification, it is not required and would not be flooded with water immediately following an event since there is a curb around it. Also, since switchover to sump recirculation is automatic, verification of water level is not required nor part of a preplanned manual safety function. The containment RHR sump level indicators have not been shown to be significant to public health and safety by either operational experience or PSA. They are not included in the WCGS IPE model. Therefore, the RHR sump level instrumentation will not be added to the TS.

8. **Containment Isolation Valve Position**

The EOPs make use of this indication as part of an immediate response to a reactor trip with safety injection actuated. The operator is directed to confirm containment isolation, for both Phase A and B signals, as an immediate response to any safety injection. If the position indication shows any valves to be open, then the operator is directed to close them. Failure of this indication or failure of the operator to manually close an open valve could result in a release path for radioactive materials to the environment. However, a double failure would have to occur which is not a DBA requirement. For DBAs, there are Type A variables (containment pressure, normal sump level, containment radiation) which provide the operator with information required to perform actions which ensure the containment integrity critical safety function during a DBA. While containment isolation is an important aspect of the containment analysis, indication of isolation valve position has not been shown to be significant to public health and safety by either operational experience or PSA. Containment isolation valve position indication is not included in the WCGS IPE. Therefore, this instrumentation will not be added to the TS.

9. **Containment Hydrogen Concentration Level**

TS 3/4.6.4, Combustible Gas Control, which requires the operability of the containment hydrogen analyzers, is evaluated on the TS Screening Form for LCO 3.6.4.1. In accordance with that form and the Safety Evaluation, Attachment 1, LCO 3.6.4.1 will be deleted, since it is redundant to LCO 3.3.3.6 and is obsolete per the STS.

10. **Radiation Level in RCS**

R.G. 1.97 defines the purpose of monitoring this variable as detection of breach (of the fuel cladding). The EOPs do not base any decisions or actions on this variable. This variable is not required to assure the integrity of any fission product barrier. This TS does not satisfy criterion 3. The radiation level in the RCS is an important indicator of a fuel cladding breach; however, it has not been shown to be significant to public health and safety by either operational experience or PSA, and thus does not satisfy criterion 4. The RCS radiation level indication has not been modeled in the WCGS IPE. For these reasons, this variable need not be incorporated into TS.

11. **Auxiliary Feedwater Flow Rate**

The auxiliary feedwater flow rate indicator has not been shown to be significant to public health and safety by either operational experience or PSA. It is not included in the WCGS IPE model. However, the AFW flow rate indication should be retained for several reasons. First, it is included in NUREG-1431 Table 3.3.3-1. Second, SR 4.7.1.2.1 requires AFW flow rate indication. Third, AFW flow rate indication is being retained in TS Table 3.3-9 for the auxiliary shutdown panel (ASP). If an LCO and SR for the ASP AFW flow rate indication are being retained, then they should also be retained in TS 3.3.3.6.

12. **PORV and PORV Block Valve Position Indicators**

PORV and PORV block valve position indicators have been deleted from Technical Specification 3.3.3.6. Loss of position indication requires that the Actions associated with LCO 3.4.4 be entered; therefore, there is no need to also have these indicators under LCO 3.3.3.6. It is further noted that these indicators are not Type A variables at Wolf Creek, nor are they RG 1.97 Category 1. Monthly channel checks for these indicators have been added as SR 4.4.4.3 and SR 4.4.4.4.

13. **Safety Valve Position Indicator**

This instrument is not a Type A or Category 1 indication. The safety valve position indicators have not been shown to be significant to public health and safety by either operational experience or PSA, and they are not included in the WCGS IPE model. This instrument is a Type D, Category 2 variable and will be relocated.

14. **Unit Vent - High Range Noble Gas Monitor**

This instrument is not a Type A or Category 1 indication. The unit vent - high range noble gas monitor has not been shown to be significant to public health and safety by either operational experience or PSA. It is not included in the WCGS IPE model. The WCGS Emergency Plan also has provisions to issue offsite Protective Action Recommendations based on plant condition. Indication from this monitor would not be required to make those determinations. This is a Type D, Category 2 variable and will be relocated.

(4) **CONCLUSION**

<u>*</u> This Technical Specification is retained.

*As indicated above; Neutron Flux and RVLIS will be added.

<u>**</u> The Technical Specification may be relocated to the following controlled document(s):

**USAR Chapter 16 (Containment pressure-Extended-range; Safety valve position indicator; Unit vent wide-range noble gas monitor).

.'ORV and PORV Block Valve position indicators have been deleted from LCO 3.3.3.6 as discussed above.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.3.3.9 LOOSE-PART MONITORING INSTRUMENTATION
Applicable Modes: 1 and 2

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The Loose Part Detection Instrumentation provides the capability to detect loose parts in the RCS which could cause damage to some component in the RCS. Loose parts are not assumed to initiate any DBA, and the detection of a loose part is not required for mitigation of any DBA.

The Loose Part Detection System TS is not associated with installed instrumentation used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The Loose Part Detection Instrumentation TS is not applicable to a process variable, design feature, or operating restriction that is an initial condition of any DBA or transient analysis. Thus, this TS does not satisfy criterion 2.

The Loose Part Detection Instrumentation TS does not apply to any SSC assumed to function in the safety analysis. It is not part of the primary success path which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 3.

From Reference 2, the Loose-Part Monitoring Instrumentation has not been shown to be significant to public health and safety by either operational experience or PSA. Loose parts would not be expected to damage the RCS pressure boundary or affect initiating event frequencies or PRA results. The Loose-Part Monitoring System is not modeled in the WCGS IPE. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16

3339.doc

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.3.3.11 EXPLOSIVE GAS MONITORING INSTRUMENTATION
Applicable Modes: During Waste Gas Holdup System operation

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The explosive gas monitoring instrumentation provides the capability to detect the concentration of oxygen and hydrogen in the waste gas holdup system (at the hydrogen recombiners) and provide an alarm if the concentrations exceed prescribed limits. According to LCO 3.3.3.11, this TS assures the operability of the instrumentation required for LCO 3.11.2.5, Explosive Gas Mixture of the Radioactive Effluents TS. According to the Bases of LCO 3.11.2.5, the purpose of the limits on explosive gas concentrations and the monitoring instrumentation is to prevent an explosion in the waste gas holdup system. (The Bases for 3.3.3.11 were deleted in Operating License Amendment No. 42.) An explosion could result in a release of radioactive materials contained in the gaseous waste holdup system. Although release of the contents of a waste gas decay tank is an analyzed DBA, the analysis assumes that the tank ruptures non-mechanistically and not as the result of a hydrogen explosion. Therefore, the explosive gas limits are not an initial condition of a DBA.

The explosive gas monitoring instrumentation is not applicable to installed instrumentation used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The explosive gas monitoring instrumentation is not applicable to a process variable, design feature, or operating restriction that is an initial condition of any DBA or transient analysis. Thus, this TS does not satisfy criterion 2.

The explosive gas monitoring instrumentation is not assumed to function in the safety analysis. It is not a part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 3.

From Reference 2, the Explosive Gas Monitoring Instrumentation has not been shown to be significant to public health and safety by either operational experience or PSA. The function of this instrumentation is to preclude inadvertent radioactivity releases from the Waste Gas Holdup System due to a tank failure from a waste gas explosion. Severe accidents dominate public risk, not inadvertent releases. This system is not modeled in the WCGS IPE. Thus, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16. (The LCO will be relocated but a program statement will be added to new TS Section 6.8.5).

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.3.4 TURBINE OVERSPEED PROTECTION
Applicable Modes: 1, 2, and 3

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition ot a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The Turbine Overspeed Protection System actuates to mitigate a potential turbine overspeed event. This prevents the generation of potentially damaging missiles from the turbine. The turbine overspeed event is not a DBA. This event is evaluated to determine the probability of damage to equipment needed for safe shutdown. The turbine has a favorable orientation from the standpoint of low trajectory missiles; however, the combination of overspeed probability with high trajectory strike probability must meet the NRC's requirements for overall probability, i.e., less than 1E-7 per year.

The Turbine Overspeed Protection System is not applicable to installed instrumentation used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The Turbine Overspeed Protection System is not associated with a process variable, design feature, or operating restriction that is an initial condition of any DBA or transient analysis. Thus, this TS does not satisfy criterion 2.

The Turbine Overspeed Protection System is not assumed to function in the safety analysis. It does not apply to any SSC that is a part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 3.

From Reference 2, the turbine overspeed protection has not been shown to be significant to public health and safety by either operational experience or PSA. PRA studies discussed in Reference 2 indicate that the probability of turbine missile ejection and resultant damage to safety-related structures are so low that they have little or no impact on the quantification of core damage frequency. Due to the physical location and orientation of the turbine at WCGS in relation to the majority of safety-related equipment, these low probability values will be valid. Turbine overspeed protection is not included in the WCGS IPE model. Thus, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16. (The LCO may be relocated but a program statement will be added to new TS Section 6.8.5).

TS3L

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.4.2.1 SAFETY VALVES - SHUTDOWN
Applicable Modes: 4 and 5

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

This TS is applicable to Modes 4 and 5. The safety valves, together with the reactor protection system, protect the RCS from being pressurized above its Safety Limit of 2735 psig. The pressurizer safety valves provide overpressure protection during both power operation and hot standby. However, the safety valves are not assumed to function to mitigate a DBA or transient in Modes 4 and 5. According to the Bases, only one safety valve is required to relieve any overpressure condition which could occur during shutdown. In the event no safety valves are operable during shutdown there are several other means to provide the required protection, for example, the RHR relief valves in an operating RHR loop connected to the RCS or the Overpressure Protection System, which relies on the Pressurizer PORVs, can provide the needed protection. Ref. 2 Bases note that overpressure protection during shutdown is provided by operating procedures and by meeting the requirements of the LCO for low temperature overpressure protection (LCO 3.4.9.3). LCO 3.4.9.3 is applicable when in Mode 3 and any RCS cold leg temperature is less than or equal to 368 °F and in Modes 4, 5, and 6 with the vessel head installed.

The safety valve TS is not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB. This TS does not satisfy criterion 1.

The safety valve TS is not associated with a process variable, design feature or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The safety valves are not assumed to function in the safety analysis to mitigate overpressure transients in Modes 4 and 5. The pressurizer safety valve TS is not applicable to components that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

From Reference 2, the operation of the pressurizer safety valves during shutdown has not been shown to be significant to public health and safety by either operational experience or PSA. A recent probabilistic study indicated that the frequency of experiencing an event leading to RCS pressure above 2485 psig while at shutdown conditions is less than 1.0E-10 per year. As the WCGS IPE model does not include shutdown conditions, the pressurizer safety valves are not in the model. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.4.5 STEAM GENERATORS
Applicable Modes: 1, 2, 3, and 4

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

This TS establishes the inservice inspection requirements for the steam generator (SG) tubes which are part of the RCPB. It is intended to maintain the structural integrity of this portion of the RCPB. The LCO requires the SGs to be operable in Modes 1, 2, 3, and 4; operability in this case refers to the structural integrity of the SG tubes by means of an augmented inservice inspection (ISI) program that is performed periodically during plant outages.

This specification is not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB; and, therefore, this TS does not satisfy criterion 1.

This specification is not applicable to a process variable or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The specification is applicable to the design feature of SG tube strength which comes into play, for example, during a LOCA or MSLB to avoid a combined LOCA/SGTR or MSLB/SGTR event. However, tube integrity is neither an active design feature nor monitored or controlled during plant operation, rather during shutdown conditions under the SG ISI program. Thus, the structural integrity and assumed passive post-accident performance of the SG tubes is maintained by periodic inspection. Therefore, this TS does not satisfy criterion 2.

The SG tubes are components of the RCS that are part of the primary success path and which function to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The post-accident or post-transient performance of the SGs, which is a passive function, is maintained by the periodic inspection and repair of the SG tubes specified in this LCO. However, the operability of the SG tubes is not maintained during operation of the plant through any actions performed or parameters monitored by the operating staff. Also, the SG tubes do not perform any active function or actuation required for DBA or transient mitigation. Therefore, this TS does not satisfy criterion 3.

Reference 2 states that the steam generators have not been shown to be significant to public health and safety by either operational experience or PSA. Also, for WCGS, the SGTR initiated event contributes less than 1.5% to the core damage frequency. WCGS does have three dominant (i.e. above 1.0E-07/year) containment bypass sequences resulting from the SGTR initiated event. They are: 1. SGTR event, AFW and cooldown fail; 2. SGTR event, failure to stabilize RCS and ruptured SG pressure, secondary side relief valve (RV) closes; and 3. SGTR event, failure to stabilize RCS and ruptured SG pressure, secondary side RV sticks open, cooldown fails. However, steam generator bypass is controlled by Technical Specification 3.4.6, "Leakage Detection Systems", which limits the leakage from all steam generators not isolated from the RCS to 1 gallon per minute. This limitation assures that dosage contribution from the tube leakage will be limited to a small fraction of the 10CFR Part 100 dose guideline values in the event of a SGTR event. Therefore, this TS does not satisfy Criterion 4.

Ref. 4 concluded that this LCO could be relocated out of TS but that the SRs must be retained.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

The LCO may be relocated to USAR Chapter 16; however, a SG tube surveillance program statement will be added to new TS Section 6.8.5.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.4.7 CHEMISTRY
Applicable Modes: At all times

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

This specification places limits on the oxygen, chloride, and fluoride content of the RCS to minimize corrosion of the RCPB.

The RCS chemistry TS is not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB. The RCS chemistry TS does not satisfy criterion 1.

Chemistry restrictions are not used as initial conditions for safety analysis. However, the chemistry requirements are applicable, albeit indirectly, to a design feature (RCS integrity) that is an initial condition of a DBA or transient analysis that either assumes the failure or presents a challenge to the integrity of a fission product barrier. But RCS integrity is a passive rather than an active design feature. Thus, the RCS chemistry TS does not satisfy criterion 2.

The chemistry requirements for the RCS are applicable to the integrity of the RCS which is a system that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, the chemistry requirements do not directly assure the RCS integrity, but provide an indication of a concern. RCS integrity is assured through ISI and engineering evaluations of structural integrity. Therefore, the RCS chemistry TS does not satisfy criterion 3.

From Reference 2, RCS chemistry has not been shown to be significant to public health and safety by either operational experience or PSA. Primary system corrosion is a slow process which would be detected in inservice inspections or small leakages before it caused a rupture. Undetected corrosion would not be expected to have a significant affect on LOCA frequencies. RCS chemistry is not modeled in the WCGS IPE. Therefore, the RCS chemistry TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.4.9.2 P/T LIMITS - PRESSURIZER
Applicable Modes: At all times

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

Pressure and temperature (P/T) limits are placed on the pressurizer (PZR) to be consistent with the requirements of the ASME Code. In accordance with the Bases, although the PZR operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operational limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The P/T limits are not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB. Therefore, this TS does not satisfy criterion 1.

The P/T limits are not applicable to a process variable or design feature that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. While the TS imposes operating restrictions, they are not associated with a DBA or transient analysis or with precluding the occurrence of an unanalyzed event but, rather, with maintaining fatigue cycles within approved limits. Therefore, this TS does not satisfy criterion 2.

The P/T limits are associated with an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. For example, the PZR must maintain its structural integrity following a MSLB or SBLOCA to maintain RCS circulation and cooling capability. However, the passive functional integrity of the PZR is not maintained by any activities of the plant staff during plant operation. Pressurizer integrity is a design feature maintained by ASME Code design and component cyclic/transient limit requirements imposed outside of this TS. Thus, this TS does not satisfy criterion 3.

From Reference 2, the Pressurizer P/T limit technical specification has not been shown to be significant to public health and safety by either operational experience or PSA. The consequences of a pressurizer failure due to operation outside the specified pressure/temperature limits are expected to be much less severe that of a reactor vessel failure, which has been shown in risk studies to not be a dominant risk. While the pressurizer P/T limits are not modeled in the WCGS IPE, they are included in the Modular Accident Analysis Program (MAAP) computer code which provides Level 1 success criteria and Level 2 releases. Thus, the TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.4.10 STRUCTURAL INTEGRITY
Applicable Modes: All Modes

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

This specification provides the inspection requirements for the ASME Code Class 1,2, and 3 components to ensure their structural integrity.

This specification is not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB. Therefore, the structural integrity requirements do not satisfy criterion 1.

This specification is not applicable to a process variable, design feature, or operating restriction that is an initial condition of DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. While the TS imposes an operating restriction regarding pressure boundary integrity, it is not monitored or controlled during plant operation. The assumed integrity of Class 1, 2, and 3 components is assured by means of periodic inspections. Therefore, this TS does not satisfy criterion 2.

ASME Code Class 1, 2, and 3 components are part of the primary success path and function to mitigate DBAs or transients that either assume the failure of or present a challenge to the integrity of a fission product barrier. Individual ASME Code Class 1, 2, and 3 components may satisfy criterion 3 and the requirements that ensure the integrity/operability of these components are included in the individual specifications that cover these components. However, as stated above, this specification addresses the passive, pressure boundary function of these components. Therefore, this TS does not satisfy criterion 3.

Ref. 4 concluded that the LCO for this specification could be relocated out of TS; however, the associated SR must be relocated to the TS programmatic requirements.

From Reference 2, the structural integrity of ASME Code 1, 2 and 3 components has not been shown to be significant to public health and safety by either operational experience or PSA. Failure modes of these components would not be identified from the requirements of this technical specification. The structural integrity of ASME Code 1, 2 and 3 components is not modeled in the WCGS IPE. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16. (The LCO may be relocated but a program statement will be added to new TS Section 6.8.5).

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.4.11 REACTOR COOLANT SYSTEM VENTS
Applicable Modes: 1, 2, 3, and 4

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The RCS vents are provided to exhaust, from the reactor vessel, noncondensable gases and/or steam from the RCS which could inhibit natural circulation core cooling following any event involving a loss of offsite power and requiring long term cooling, such as a LOCA. Their function, capabilities, and testing requirements are consistent with NUREG-0737, Item II.B.1, which assumes a severely damaged core. However, the vents are not required to operate to mitigate any DBA or transient. Operation of the vents is not assumed in the safety analysis. This is because operation of the vents is not part of the primary success path. Operation of the vents is an assumed operator action after an event has occurred and is required only if there is indication that natural circulation is not occurring.

The TS requirements for RCS vents are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The RCS vents TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The TS for the RCS vents does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, the RCS vent requirements do not satisfy criterion 3.

From Reference 2, the reactor coolant system vents have not been shown to be significant to public health and safety by either operational experience or PSA. On Westinghouse designed PWRs, buildup of sufficient non-condensable gases or steam within the primary system to inhibit natural circulation is unlikely. Also, the contribution of inadvertent opening of the head vent valves to the small LOCA initiating event frequency is not a primary contributor to risk. The reactor coolant system vents are not modeled in the WCGS IPE. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is relocated.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.6.1.2 CONTAINMENT LEAKAGE
Applicable Modes: 1, 2, 3, and 4

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

This TS identifies the allowable leakage rates for the containment structure which are established to meet 10 CFR 50, Appendix J. These requirements ensure that the leakage rates from containment will not exceed the value assumed in the safety analyses at the peak accident pressure.

This specification is not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of a the RCPB, and, therefore, the TS does not satisfy criterion 1.

This specification is applicable to parameters that are an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, the process variables for which the requirements are applicable (containment design pressure and allowable leakage rates) are not variables that are monitored and controlled during power operation such that process values remain within the analysis bounds. Containment integrity is assured by periodic inspection and testing. Therefore, this specification does not satisfy criterion 2.

The specification applies to containment leakage rate limits. Thus, it is applicable to a structure that is part of the primary success path and which function to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, the intent of criterion 3 is to capture only those SSC (and supporting systems) that are part of the primary success path of a safety sequence analysis. Operability of the containment is assured by a separate LCO (3.6.1.1), and the limits imposed by the leakage rate requirements are neither monitored or controlled during operation nor part of the primary success path of the containment function. Therefore, this TS does not satisfy criterion 3.

From Reference 2, containment leakage has not been shown to be significant to public health and safety by either operational experience or PSA. PRAs indicate that risk is dominated by events in which the containment is bypassed, unisolated, or fails structurally. The technical specification value for overall containment leakage is included in the WCGS MAAP model, but contributes only a small fraction of the total release in the Level 2 IPE. Therefore, this TS does not satisfy criterion 4.

Ref. 4 concluded that this LCO could be relocated out of TS but that the limiting values of P_a and L_a must be retained in TS.

(4) **CONCLUSION**

_ This Technical Specification is retained.

* The Technical Specification may be relocated to the following controlled document(s):

* The LCO may be relocated to USAR Chapter 16, but the limiting values of P_a and L_a will be retained in the Containment Integrity Bases. Relocation of the LCO requires that revisions be made to SR 4.6.1.1c and SR 4.6.1.7.2.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY
Applicable Modes: 1, 2, 3, and 4

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The containment serves as a barrier to prevent the release of fission products following a LOCA or MSLB inside containment. To mitigate the potential consequences of a DBA, it is necessary that the containment structure meet its structural requirements. This specification is intended to detect abnormal degradation of the containment structural elements. This TS outlines an appropriate inspection and testing program to demonstrate this capability. The program consists of the measurement of tendon liftoff force, tensile tests of tendon wires, and visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment.

This specification is not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of a the RCPB; and, therefore, this TS does not satisfy criterion 1.

This specification is applicable to a design feature (the containment) that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Containment structural integrity is assumed to be available for many DBAs. However, containment structural integrity is not monitored or controlled during plant operation but, rather, via periodic inspections and tests. Therefore, this TS does not satisfy criterion 2.

The specification applies to the detection of abnormal degradation of containment structures and therefore to containment structural integrity. Thus, it is applicable to a structure that is part of the primary success path which functions to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, the functional mode addressed by the TS is maintaining the passive, pressure boundary integrity. This TS does not address the capability of the containment to function or actuate in order to mitigate the consequences of a DBA or transient. Therefore, this TS is not required to ensure the operability of containment and, thus, does not satisfy criterion 3.

Ref. 4 concluded that this LCO could be relocated out of TS but that the associated SRs should be retained to meet the operability requirements for a retained LCO, in this case LCO 3.6.1.1. Ref. 2 incorporated the SRs regarding tendon surveillance into Section 6 of the TS.

From Reference 2, containment leakage has not been shown to be significant to public health and safety by either operational experience or PSA. PRAs indicate that risk is dominated by events in which the containment is bypassed, unisolated, or fails structurally. None of the sequences addressed in the containment and source term analysis could realistically threaten containment due to hydrogen combustion. No WCGS containment vulnerabilities were identified as a result of Supplement 3 to Generic Letter 88-20.

The WCGS IPE also showed that the overall release frequency per year is dominated by releases due to containment bypass sequences of which interfacing system LOCAs make up the vast majority.

The best estimate containment failure mode will occur at 2.13 times the design pressure due to membrane stresses in the containment mid-height region which exceed the pre-stress and cause through-concrete cracking and yielding of the liner, reinforcing steel, and pre-stressing tendons. The material properties used in these calculations do not change rapidly, so testing and inspection requirements of this technical specification are not critical. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16. (The LCO may be relocated; but a program statement will be added to new TS Section 6.8.5).

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.6.4.1 HYDROGEN ANALYZERS

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier
X	_	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) DISCUSSION

The operability of the systems and equipment required for the detection and 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. The hydrogen monitoring system is initiated after the occurrence of a LOCA to detect the buildup of hydrogen. The purpose of the hydrogen control features is to maintain containment integrity by assuring that a hydrogen burn or explosion would not overpressurize containment.

The TS requirements for hydrogen analyzers are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The hydrogen analyzers TS are not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this requirement does not meet criterion 2.

The TS for hydrogen analyzers are applicable to an SSC (containment) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, these requirements satisfy criterion 3.

LCO 3.6.4.1 is deleted since it is redundant to LCO 3.3.3.6 and is obsolete per the STS.

(4) CONCLUSION

X The requirements of this Technical Specification are retained under LCO 3.3.3.6..

_ The Technical Specification may be relocated to the following controlled document(s):

X This Technical Specification is deleted.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.7.2 STEAM GENERATOR P/T LIMITATION
Applicable Modes: At all times

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
__	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
__	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
__	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

Pressure and temperature (P/T) limits are placed on the steam generators (SG) to prevent a non-ductile failure of either the RCPB or the secondary side pressure boundary. The specification places limits on the SG P/T to ensure that the pressure induced stresses are within the maximum allowable fracture toughness stress limits. The P/T limits are based on a SG RT_{NDT} sufficient to prevent brittle fracture.

The SG P/T limits are not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB. Therefore, the SG P/T limits do not satisfy criterion 1.

The P/T limits are not applicable to a process variable, design feature, or operating restriction that is an initial condition of DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. While the TS imposes an operating restriction, it is not employed to prevent unanalyzed accidents and transients. Under the conditions when this TS could be required, an unanalyzed event of any significance from a safety function standpoint (decay heat removal, accident mitigation, and reactor shutdown) is unlikely to result. Therefore, this TS does not satisfy criterion 2.

The P/T limits are associated with an SSC that is part of the primary success path which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. For example, the SG must maintain its structural integrity following a MSLB or SBLOCA to maintain RCS circulation and cooling capability. However, the TS limitations apply only to shutdown conditions when RCS temperature is unusually low (less than 70 °F). Under these conditions, the SG is not required to function to mitigate any DBAs or transients. Therefore, this TS does not satisfy criterion 3.

From Reference 2, the steam generator pressure/temperature limitation has not been shown to be significant to public health and safety by either operational experience or PSA. This technical specification is intended to prevent brittle fracture of a SG when at low pressures and temperatures, something which is not likely during plant operation, which is the analyzed condition for the WCGS IPE study. This condition, then, is not modeled in the WCGS IPE. Therefore, the TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

372.doc

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.7.8 SNUBBERS

Applicable Modes: 1, 2, 3, and 4. Also Modes 5 and 6 for those systems required to be operable in these Modes.

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The snubbers are required to be operable to ensure that the structural integrity of the RCS and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. The restraining action of the snubbers ensures that the initiating event failure does not propagate to other parts of the failed system or to other safety systems. Snubbers also allow normal thermal expansion of piping and nozzles to eliminate excessive thermal stresses during heatup or cooldown. Snubber surveillance is conducted under the requirements of the Wolf Creek Snubber Surveillance Program.

The TS requirements for snubbers are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The snubber TS is associated with a design feature or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, the snubber requirements are not explicitly considered in the accident analysis. The availability of the snubbers is assumed based on the performance of a program of periodic augmented inspection and testing. Snubber operability is not required to be monitored and controlled during plant operation. Some snubbers (inaccessible) can only be inspected during plant outages. Thus, this TS does not satisfy criterion 2.

Those snubbers that are required to function during DBAs or transients to prevent the initiating event from propagating to other systems or components that are part of the primary success path may be considered components that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, snubbers are not explicitly considered in DBA or transient analyses but are a structural/design feature whose operability is assured by an inspection program. Therefore, this TS does not satisfy criterion 3.

From Reference 2, for non-RCS and other high energy systems such as the feedwater and main steam systems inside the containment building, the snubber technical specification has not been shown to be significant to public health and safety by either operational experience or PSA. Reference 2 reviewed the Zion and Millstone PRAs and determined that the snubbers, which ensure the operability of certain safety-related equipment during a seismic event, are not risk dominant for this scenario. While the seismic portion of the WCGS IPEEE is not complete at this time, there is no reason to believe the results will be different. Thus, this TS does not satisfy criterion 4.

For snubbers which are not part of the RCS or other high energy systems, then, this technical specification can be relocated.

(4) **CONCLUSION**

__ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16. (The LCO may be relocated but a program statement will be added to new TS Section 6.8.5).

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.7.9 SEALED SOURCE CONTAMINATION

Applicable Modes: At all times

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The TS limitations ensures that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR Part 70.39(a)(3) limits for plutonium.

The TS requirements for sealed source contamination are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The sealed source contamination TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The TS for sealed source contamination does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

From Reference 2, sealed source contamination has not been shown to be significant to public health and safety by either operational experience or PSA. Sealed sources are used for calibration and other purposes which have no impact on plant risk. This technical specification is not included in the WCGS IPE. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.7.12 AREA TEMPERATURE MONITORING
Applicable Modes: Whenever equipment in the area is required to be OPERABLE.

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

This specification places a limit on the temperature of the areas of the plant which contain safety-related equipment. This is required to ensure that the temperature of the equipment does not exceed its environmental qualification temperature during normal operation. Exposure to excessively high temperatures may degrade the equipment and cause a loss of its operability.

The TS requirements for area temperature monitoring are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The area temperature monitoring TS is associated with the variable of room temperature which is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The TS for area temperature monitoring does apply to the operability of SSCs that are part of the primary success path which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, the TS is only indirectly applicable to the operability of these systems and components. Therefore, this TS does not satisfy criterion 3.

From Reference 2, the area temperature monitors have not been shown to be significant to public health and safety by either operational experience or PSA. The area temperature monitors have not been included in the WCGS IPE. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16. (The LCO may be relocated but a program statement will be added to new TS Section 6.8.5).

3712.doc

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.8.4.1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Applicable Modes: 1, 2, 3, and 4

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The containment penetration conductor overcurrent protective devices are installed to minimize the potential for a fault in a component inside containment, or in cabling which penetrates containment. This prevents an electrical penetration from being damaged in such a way that the containment structure is breached.

The TS requirements for these devices are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The containment penetration conductor overcurrent protective devices do help to preserve the assumptions of the accident analysis by enhancing proper equipment operation. However, they are not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The containment penetration conductor overcurrent protective devices provide equipment and distribution system protection from faults or improper operation of other protective devices in addition to that provided by the design of the distribution system. The TS for containment penetration conductor overcurrent protective devices does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

The containment penetration conductor overcurrent protective devices are installed to minimize the potential for a fault in a component inside containment or in cabling which penetrates the containment from damaging the electrical penetration in such a way that the containment structure is breached. From Reference 2, the protective devices have not been shown to be significant to public health and safety by either operational experience or PSA. The overcurrent protective devices have not been included in the WCGS IPE. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

3841.doc

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.9.5 COMMUNICATIONS

Applicable Modes: During Core Alterations

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) DISCUSSION

This specification requires communication between the control room and the refueling station to ensure that any abnormal change in the facility status or core reactivity observed on the control room instrumentation can be communicated to the refueling station personnel during core alterations.

The TS requirements for communications are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The communications TS are not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The TS for refueling communications does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

From Reference 2, communications between the control room and the refueling station during core modifications has not been shown to be significant to public health and safety by either operational experience or PSA. The WCGS IPE does not model the plant during refueling operations. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

395.doc

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.9.6 REFUELING MACHINE

Applicable Modes: During movement of drive rods or fuel assemblies within the Reactor Vessel.

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

This specification assures that: (1) manipulator cranes will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

The TS requirements for the refueling machine are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The refueling machine TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The TS for the refueling machine does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

The requirements of this technical specification are not significant to public health and safety by either operational experience or PSA. The refueling machine is used to transport fuel assemblies during refueling operations. The WCGS IPE models the plant during power operations, and therefore does not include the refueling machine in any risk quantifications. However, if the refueling machine were included in the model, it's significance would negligible. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

396.doc

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY
Applicable Modes: With fuel assemblies in the spent fuel storage facility.

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

This specification ensures that loads in excess of one fuel assembly containing a control rod, plus the weight of the fuel handling tool, will not be moved over other fuel assemblies stored on the spent fuel storage racks. Therefore, in the event of a drop of this load, the activity released is limited to that contained in one fuel assembly. This also prevents any possible distortion of fuel assemblies in the storage racks from achieving a critical configuration. This specification applies to prevention of a heavy load drop accident and assures that the damage caused by the load is limited to the equivalent of one spent fuel assembly. This assumption is consistent with the activity release assumed in the DBA accident analyses for a fuel handling accident; however, the load drop event is not a DBA.

The TS requirements for crane travel are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The spent fuel facility crane travel TS is associated with an operating restriction for a heavy load drop event. This specification is not applicable to a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The TS for crane travel does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

From Reference 2, the spent fuel storage facility crane has not been shown to be significant to public health and safety by either operational experience or PSA. Reference 2 reviewed several environmental reports related to these cranes, and found their risk significance to be minimal. The spent fuel storage facility crane is not modeled in the WCGS IPE. Therefore, this TS does not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

397.doc

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.9.10.2 WATER LEVEL - REACTOR VESSEL/CONTROL RODS
Applicable Modes: 6, during movement of control rods within the Reactor Vessel.

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

This specification places a lower limit on the amount of water above the top of the fuel assemblies in the reactor vessel during movement of control rods. The Bases state that this ensures the water removes 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly in the event of a fuel handling accident (FHA). However, the movement of control rods is not associated with the initial conditions of an FHA, and the Bases do not address any concerns regarding inadvertent criticality which could lead to a breach of the fuel rod cladding. Inadvertent criticality during Mode 6 is prevented by maintaining proper boron concentration in the coolant in accordance with LCO 3.9.1.

The TS requirements for water level - reactor vessel are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The wate... el - reactor vessel TS is not associated with a process variable or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The TS for water level - reactor vessel do not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

The reactor water level during refueling operations has not been shown to be significant to public health and safety by either operational experience or PSA. While refueling operations, including the reactor water level, are not modeled in the WCGS IPE, but they would not be important in any of the dominant accident sequences at WCGS.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.10.1 SPECIAL TEST EXCEPTION - SHUTDOWN MARGIN
Applicable Mode: 2

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable , design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

Ref. 4 states that "Special Test Exceptions 3.10.1 through 3.10.4 may be included with corresponding LCOs which are remaining in Technical Specifications. Special Test Exception 3.10.5 may be relocated outside of Technical Specifications along with LCO 3.1.3.3."

LCO 3.10.1 is only applicable in Mode 2. As discussed in the Screening Form for TS 3.1.1.1, the SDM requirements for Modes 1 and 2 are retained in other Reactivity Control System Technical Specifications. Retained Special Test Exceptions 3.10.2 and 3.10.3 address Special Test Exception 3.10.1 for LCOs 3.1.3.1 and 3.1.3.6. Therefore, Technical Specifications 3.10.1 will be deleted.

Shutdown margin has been shown to not be a dominant risk contributor. See TS 3.1.1.1. Therefore, these requirements do not satisfy criterion 4.

(4) **CONCLUSION**

_ This Technical Specification is retained.

_ The Technical Specification may be relocated to the following controlled document(s):

X This Technical Specification is deleted.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.10.5 SPECIAL TEST EXCEPTION - POSITION INDICATION SYSTEM - SHUTDOWN

Applicable Modes: 3, 4, and 5 during performance of rod drop time measurements and during surveillance of DRPI for Operability.

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable , design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

Ref. 4 states that "Special Test Exceptions 3.10.1 through 3.10.4 may be included with corresponding LCOs which are remaining in Technical Specifications. Special Test Exception 3.10.5 may be relocated outside of Technical Specifications along with LCO 3.1.3.3."

In accordance with its Screening Form, LCO 3.1.3.3 may be relocated from TS. Therefore, LCO 3.10.5 may be relocated.

(4) **CONCLUSION**

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16.

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.11.1.4 LIQUID HOLDUP TANKS
Applicable Modes: At all times

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The liquid holdup tank specifications impose limits on the quantity of radioactive material contained in specific outdoor tanks that may contain radwaste. Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentration would be 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. The tanks addressed by this specification are:

a. Reactor Makeup Water Storage Tank
b. Refueling Water Storage Tank
c. Condensate Storage Tank
d. Outside temporary tanks, excluding demineralizer vessels and liners being used to solidify radioactive wastes.

These tanks are not addressed by the safety analysis of radioactive release from a subsystem or component.

The TS requirements for liquid holdup tanks are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The liquid holdup tanks TS is not associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The TS for liquid holdup do not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

From Reference 2, the liquid holdup tanks, which hold radwaste, have not been shown to be significant to public health and safety by either operational experience or PSA. Risk of radioactivity release is dominated by severe accidents, not releases of radionuclides generated from normal operations. For this reason, the liquid holdup tanks are not modeled in the WCGS IPE. Therefore, this TS do not satisfy criterion 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16. (The LCO may be relocated but a program statement will be added to new TS Section 6.8.5).

TECHNICAL SPECIFICATION SCREENING FORM

(1) TECHNICAL SPECIFICATION 3.11.2.5 EXPLOSIVE GAS MIXTURE
Applicable Modes: At all times

(2) EVALUATION OF POLICY STATEMENT CRITERIA

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be included in the new Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) DISCUSSION

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining these limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of GDC 60 of Appendix A to 10 CFR 50. The accident analysis concerning the gaseous radwaste system assumes that a storage tank ruptures, from unspecified causes, and releases its contents without mitigation.

The TS requirements for explosive gas mixture are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The explosive gas mixture TS is associated with a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Thus, this TS does not satisfy criterion 2.

The TS for explosive gas mixture does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS do not satisfy criterion 3.

The explosive gas mixture of the waste gas holdup tanks has not been shown to be significant to public health and safety by either operational experience or PSA. Risk of radioactivity release is dominated by severe accidents, not releases of radionuclides generated from normal operations. In addition, from Reference 2 the quantity of radioactivity contained in each pressurized gas storage tank in the waste gas holdup system is limited to assure a release would be substantially below the dose guideline values of 10 CFR Part 100. The waste gas holdup tanks are not modeled in the WCGS IPE. Therefore, this TS does not satisfy criterion 4.

(4) CONCLUSION

__ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16. (The LCO may be relocated but a program statement will be added to new TS Section 6.8.5).

TECHNICAL SPECIFICATION SCREENING FORM

(1) **TECHNICAL SPECIFICATION** 3.11.2.6 GAS STORAGE TANKS
Applicable Modes: At all times

(2) **EVALUATION OF POLICY STATEMENT CRITERIA**

Is the Technical Specification applicable to:

YES	NO		
_	X	(1)	Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
_	X	(2)	A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(3)	A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
_	X	(4)	A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

If the answer to any one of the above questions is "YES", then the Technical Specification shall be retained in the Technical Specifications.

If the answer to all four of the above questions is "NO", the Technical Specification may be relocated to a controlled document.

(3) **DISCUSSION**

The gas storage tank specifications impose limits on the quantity of radioactive material contained in those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another TS. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a member of the public at the nearest site boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure." The accident analysis concerning the gaseous radwaste system assumes a rupture of a storage tank without mitigation.

The TS requirements for gas storage tanks are not applicable to installed instrumentation used to detect a significant abnormal degradation of the RCPB; therefore, this TS does not satisfy criterion 1.

The gas storage tank TS is associated with a process variable or operating restriction (quantity of contained radioactivity) that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. However, the barrier in this case is the tank itself which is not a barrier that is monitored and controlled during power operation of the plant. Therefore, this TS does not satisfy criterion 2.

The TS for gas storage tanks does not apply to an SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, this TS does not satisfy criterion 3.

From Reference 2, the waste gas holdup tanks, which hold radwaste, have not been shown to be significant to public health and safety by either operational experience or PSA. Risk of radioactivity release is dominated by severe accidents, not releases of radionuclides generated from normal operations. In addition, from Reference 2 the quantity of radioactivity contained in each pressurized gas storage tank in the waste gas holdup system is limited to assure a release would be substantially below the dose guideline values of 10 CFR Part 100. The waste gas holdup tanks are not modeled in the WCGS IPE. Therefore, this TS does not satisfy criterion 4.

(4) CONCLUSION

_ This Technical Specification is retained.

X The Technical Specification may be relocated to the following controlled document(s):

USAR Chapter 16. (The LCO may be relocated but a program statement will be added to new TS Section 6.8.5).

3-11-2-6

APPENDIX A

1.0 OBJECTIVE

Four criteria are included in the NRC's Final Policy Statement for determining the requirements to be included in the Technical Specifications. The Wolf Creek Technical Specifications have been evaluated based on those four criteria. The purpose of this document is to determine if the parameters, components, or systems addressed by the Technical Specifications are significant from an operating experience or probabilistic safety assessment (PSA) perspective (i.e. the fourth criterion).

2.0 EVALUATION BASES

The evaluation of the risk impact of the Technical Specifications in regards to the fourth criterion is based on the following:

A. The Technical Specifications that are relocated will be transferred to USAR Chapter 16.3 and will be implemented by programs and procedures subject to control by WCNOC, within the constraints of 10CFR50.59.

B. The risk criteria used in determining the disposition of a Technical Specification are the following:

   1. If the Technical Specification contains constraints of prime importance in limiting the likelihood or severity of the accident sequences that are found to dominate risk, it will be retained.

   2. If the Technical Specification includes items involved in one of these dominant sequences but has an insignificant impact on the probability or severity of that sequence and is not significant based on operating experience, it will be relocated to USAR Chapter 16.3.

   3. If the Technical Specification is not involved in risk dominant sequences and is not significant based on operating experience, it will be relocated to USAR Chapter 16.3.

C. The measures related to risk used in this evaluation are core damage frequency (CDF) and offsite health effects. These measures are consistent with the Final Policy Statement on Technical Specification Improvements and the Safety Goal and Severe Accident Policy Statements.

D. The criteria used to determine if a sequence is risk dominant is the following: For core melt, any sequence whose frequency was found to be greater than 1.0E-06 per reactor year in the Wolf Creek IPE (Reference 9) is considered to be a dominant sequence. This is slightly over 2% of the total Wolf Creek core damage frequency of 4.2815E-05 per reactor year. These sequences, 14 in all, are identified in Table 3.4-1 of Reference 9 and repeated here in Table 4.1. In addition, any sequence whose frequency of containment bypass was found to be greater than 1.0E-07 per reactor year in the Wolf Creek IPE is also considered to be a dominant sequence. Three sequences met this criterion and are identified in Table 3.4-4 of Reference 9 and also repeated below in Table 4.2

For offsite health effects, any sequence whose frequency of severe radioactive release was found to be greater than 1.0E-07 per reactor year in the Wolf Creek IPE is considered to be a dominant sequence. Severe radioactive release for WCGS is defined as a WASH-1400 (Reference 10) PWR release category 4 accident. As noted in Section 4.3.3.1 of Reference 9, while sequences with frequencies greater than 1.0E-07 per reactor year exist at WCGS, none of these sequences meet the PWR release category 4 inventory release criteria. Offsite health effects, then, are not a factor in determining whether to relocate the WCGS Technical Specifications.

E. Wolf Creek systems and functions that are important from a PSA or operating experience perspective are listed in Table 4.3. These identified systems, as well as the risk dominant sequences as determined in paragraph D, were used to screen the requirements of the Technical Specifications reviewed. If the requirements of a Technical Specification were not found to be modeled in the Wolf Creek IPE and no significant risk issues were identified from a review of the risk insights or operating experience, that Technical Specification would be relocated to USAR Chapter 16.3 unless the other three criteria mandated that it be retained.

3.0 METHOD USED

Screening forms were developed which formalized the review of each Technical Specification under the four criteria of the Final Policy Statement. These screening forms contain:

1. The number and title of the Technical Specification;

2. An evaluation of the Technical Specification against the Final Policy Statement's four criteria;

3. A discussion of the information used in arriving at the conclusions for the four criteria; and

4. A conclusion as to whether the Technical Specification should be retained or relocated.

This methodology is based on the approach presented in WCAP-11618 (Reference 3).

4.0 RISK DOMINANT SEQUENCES

The tables that follow contain the dominant sequences in regards to risk and containment bypass. Recall that significant offsite health effects are not a concern at WCGS.

TABLE 4.1
LIST OF DOMINANT SEQUENCES

NUMBER	SEQUENCE PROBABILITY	PERCENT CONTRIB	SEQUENCE DESCRIPTION	SEQUENCE IDENTIFIER
1	5.89E-06	13.77	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			SBO CUTSETS - COMPONENTS FAIL AFTER LSP, SBO EVENT RESULTS	SYS-CCWS
			AC POWER IS NOT RECOVERED WITHIN 8 HOURS AFTER AN SBO	8HR-FAILS
2	4.47E-06	10.44	CONTROL BUILDING SWITCHGEAR ROOMS FLOODING IEV OCCURS	IEV-FL4
3	2.87E-06	6.70	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			SBO CUTSETS - COMPONENTS FAIL AFTER LSP, SBO EVENT RESULTS	SYS-CCWS
			AC POWER IS RECOVERED WITHIN 8 HOURS AFTER AN SBO	8HR-SUCCESSFUL
			CORE UNCOVERY OCCURS WITHIN 8 HOURS AFTER AN SBO W/ RCD	CNU8-FAILS
4	2.86E-06	6.67	LOSS OF OFFSITE POWER INITIATING EVENT OCCURS	IEV-LSP
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S FROM 1/3 PUMPS	SYS-AF2WO
			OPERATOR BLEED AND FEED COOLING FAILS	SYS-OFC
			BOTH COMPONENT COOLING WATER TRAINS DO NOT FAIL	DEL-CCW
5	2.77E-06	6.48	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			SBO CUTSETS - COMPONENTS FAIL AFTER LSP, SBO EVENT RESULTS	SYS-CCWS
			AC POWER IS RECOVERED WITHIN 8 HOURS AFTER AN SBO	8HR-SUCCESSFUL
			CORE DOES NOT UNCOVER WITHIN 8 HOURS AFTER AN SBO W/ RCD	CNU8-SUCCESSFUL
			FRACTION OF AC RECOVERY AT 8 HR AFTER SBO FROM OFFSITE W/ RCD	OP-08
			HIGH PRESSURE RECIRCULATION FAILS - 1/4 PMPS FROM 1/2 TRAINS	SYS-HPR12
6	2.20E-06	5.14	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO
			COMPONENT FAIL AFTER LSP, SBO EVENT RESULTS	OTH-CCWS
			AUXILIARY FEEDWATER SYSTEM FAILS - 2/4 SG'S WITH TDAFW PUMP	SYS-AFT
			AC POWER IS RECOVERED WITHIN 2 HOURS AFTER AN SBO	2HR-SUCCESSFUL
			CORE DOES NOT UNCOVER WITHIN 2 HOURS AFTER AN SBO W/O RCD	CNU2F-SUCCESSFUL
			FRACTION OF AC RECOVERY AT 2 HR AFTER SBO FROM EDG W/O RCD	EDF-02
			HIGH PRESSURE SI RESTORATION AFTER SBO, SWS OR CCW FAILS 2/2	OTH-RRI22
7	2.20E-06	5.13	RECOVERABLE CONTROL BUILDING BASEMENT FLOOD IEV OCCURS	IEV-FL3B
			FAILURE TO MITIGATE THE CONTROL BUILDING FLOOD EVENT	SYS-FL3B
8	2.19E-06	5.11	LOSS OF THE OPERATING CCW TRAIN IEV OCCURS	IEV-CCWA
			HRA FAILURE TO PROVIDE RCP SEAL COOL IN TIMELY MANNER	OPA-RCPSEAL
			HRA SUCCESS TO TRIP RUNNING RCP ON LOSS OF SEAL COOL B4 MLO	RCPTRIP-SUC
			RCP SEAL LOCA (SLO) OCCURS AFTER CCWA LOSS (SEAL COOL LOSS)	SYS-SLOCCW
			COMPONENT COOLING WATER SYSTEM TRAIN B DOES NOT FAIL	DEL-CCWBO
9	1.86E-06	4.35	MEDIUM LOCA INITIATING EVENT OCCURS	IEV-MLO
			HIGH PRESSURE RECIRCULATION FAILS - 1/4 PMPS FROM 1/2 TRAINS	SYS-LC2
10	1.65E-06	3.84	STATION BLACKOUT INITIATING EVENT OCCURS	IEV-SBO

			SBO CUTSETS -	COMPONENTS FAIL	AFTER LSP, SBO	EVENT RESULTS	SYS-CCWS
			AUXILIARY	FEEDWATER SYSTEM	FAILS - 2/4 SG'S	WITH TDAFW PUMP	SYS-AFT
			AC POWER IS NOT	RECOVERED WITHIN	2 HOURS AFTER	AN SBO	2HR-FAILS
11	1.64E-06	3.83	STATION	BLACKOUT	INITIATING EVENT	OCCURS	IEV-SBO
			COMPONENT FAIL	AFTER LSP, SBO	EVENT RESULTS		OTH-CCWS
			AC POWER IS	RECOVERED WITHIN	8 HOURS AFTER	AN SBO	8HR-SUCCESSFUL
			CORE DOES NOT	UNCOVER WITHIN	8 HOURS AFTER	AN SBO W/ RCD	CNU8-SUCCESSFUL
			FRACTION OF AC	RECOVERY AT 8 HR	AFTER SBO FROM	FDG W/ RCD	ED-08
			HIGH PRESSURE	RECIRCULATION	FAILS - 1/2 PMPS	FROM 1/1 TRAIN	SYS-HPR11
12	1.53E-06	3.56	STATION	BLACKOUT	INITIATING EVENT	OCCURS	IEV-SBO
			SBO CUTSETS -	COMPONENTS FAIL	AFTER LSP, SBO	EVENT RESULTS	SYS-CCWS
			AUXILIARY	FEEDWATER SYSTEM	FAILS - 2/4 SG'S	WITH TDAFW PUMP	SYS-AFT
			AC POWER IS	RECOVERED WITHIN	2 HOURS AFTER	AN SBO	2HR-SUCCESSFUL
			CORE DOES NOT	UNCOVER WITHIN	2 HOURS AFTER	AN SBO W/O RCD	CNU2F-SUCCESSFUL
			FRACTION OF AC	RECOVERY AT 2 HR	AFTER SBO FROM	OFFSITE W/O RCD	OPF-02
			HIGH PRESSURE	RECIRCULATION	FAILS - 2/2 CCPS	FROM 2/2 TRAINS	SYS-HPR22
13	1.31E-06	3.06	LOSS OF OFFSITE	POWER	INITIATING EVENT	OCCURS	IEV-LSP
			COMPONENT	COOLING WATER	SYSTEM TRAIN A	FAILS	SYS-CCWA
			COMPONENT	COOLING WATER	SYSTEM TRAIN B	DOES NOT FAIL	DEL-CCWB
			HRA FAILURE TO	PROVIDE RCP SEAL	COOL IN TIMELY	MANNER	OPA-RCPSEAL
			RCP SEAL LOCA	(SLO) OCCURS -	LSP AND CCWA FLS	(SEAL COOL LOSS)	SYS-SLOLSP
14	1.15E-06	2.68	LARGE LOCA	INITIATING EVENT	OCCURS		IEV-LLO
			LOW PRESSURE	RECIRCULATION	SYSTEM FAILS		SYS-LC1

TABLE 4.2
DOMINANT CONTAINMENT BYPASS SEQUENCES

Sequence Number	Frequency	Percent Contribution (to core melt)	Sequence Description
26	2.44E-07	0.57	SGTR event, AFW and cooldown fail
27	2.41E-07	0.56	SGTR event, failure to stabilize RCS and ruptured SG pressure, secondary side RV closes
32	1.45E-07	0.34	SGTR event, failure to stabilize RCS and ruptured SG pressure, secondary side RV sticks open, cooldown fails

TABLE 4.3
SYSTEMS AND FUNCTIONS THAT ARE IMPORTANT FROM A PSA OR OPERATING EXPERIENCE PERSPECTIVE

<u>Systems/equipment generally the same as Callaway</u>
Class 1E Lower Medium Voltage System - 4.16Kv (NB)
Standby Generation System (NE)
Class 1E Low Voltage System - 480V (NG)
Class 1E 125VDC System (NK)
Class 1E Instrument AC System (NN)
Reactor Coolant Pump Seals (BB)
Auxiliary Feedwater System (AL)
Component Cooling Water System (EG)
Safety Injection System (EM)
Chemical Volume and Control System (BG)
Essential Service Water System (EF)
Residual Heat Removal System (EJ)
Pressurizer PORV's (BB)
Refueling Water Storage System (BN)
Steam Generator Atmospheric Relief Valves (??)
Containment Spray System (EN)
Solid State Protection System (SB)
7300 Process Protection System (SB)
Engineered Safety Features Actuation System (SA)
LOCA and Shutdown Sequencers (NF)
Offsite Power (NA)
Service Water System (EA)
Main Feedwater System (AE)
Auxiliary Building HVAC (for RHR) (GL)
Miscellaneous Buildings HVAC (for MDAFW) (GF)
ESW Pumphouse Ventilation System (GD)
Diesel Generator HVAC System (GM)
Emergency Fuel Oil System (JE)
Auxiliary Turbine System (FC)
Instrument Air System (KA)

ATTACHMENT VI

PROPOSED UPDATED SAFETY ANALYSIS REPORT REVISIONS

NOTE: The following USAR mark-up pages are provided as information-only pages to show WCNOC's intent to relocate the applicable portions of the Technical Specifications to the USAR. The actual page format, content, and pagination of the USAR Revision may differ slightly from the following pages. This is based on WCNOC's intent to make the new USAR Section 16 format, content, and pagination consistent with the rest of the USAR.

16.1 (3/4.1) REACTIVITY CONTROL SYSTEMS

16.1.1 INTENTIONALLY BLANK

16.1.2 (3/4.1.2) BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

16.1.2.1 (3.1.2.1) As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System if the Boric Acid Storage System is OPERABLE as given in Section 16.1.2.5a. for MODES 5 and 6 or as given in Section 16.1.2.6a for MODE 4; or

b. The flow path from the refueling water storage tank via a centrifugal charging pump to the Reactor Coolant System if the refueling water storage tank is OPERABLE as given in Section 16.1.2.5b for MODES 5 and 6 or as given in Section 16.1.2.6b for MODE 4.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

16.1.2.1.1 (4.1.2.1) At least one of the above required flow paths shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

BASES

16.1.2.1.2 The Boration Systems ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) centrifugal charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature equal to or greater than 350°F a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 17,658 gallons of 7000 ppm borated water from the boric acid storage tanks or 83,754 gallons of 2400 ppm borated water from the RWST. With the RCS average temperature less than 350°, only one boron injection flow path is required.

With the RCS temperature below 200°F, one Boration System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable in MODES 4, 5, and 6 provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or an RHR suction relief valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1.3% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2968 gallons of 7000 ppm borated water from the boric acid storage tanks or 14,071 gallons of 2400 ppm borated water from the RWST.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. In the case of the boric acid tanks, all of the contained volume is considered usable. The required usable volume may be contained in either or both of the boric acid tanks.

The limits on contained water volume and boron concentration of the RWST also 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 OPERABILITY of one Boration System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

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.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

16.1.2.2 (3.1.2.2) At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the Boric Acid Storage System via a boric acid transfer pump and a centrifugal charging pump to the Reactor Coolant System, and

b. Two flow paths from the refueling water storage tank via centrifugal charging pumps to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.3% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two flow Paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

16.1.2.2.1 (4.1.2.2) At least two of the above required flow paths shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal; and

c. At least once per 18 months by verifying that the flow path required by Section 16.1.2.2a delivers at least 30 gpm to the Reactor Coolant System.

BASES

16.1.2.2.2 See Section 16.1.2.1.2

*The provisions of Technical Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Technical Specification 4.5.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

16.1.2.3 (3.1.2.3) One centrifugal charging pump in the boron injection flow path required by Section 16.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 4, 5, and 6.

ACTION:

With no centrifugal charging pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

16.1.2.3.1 (4.1.2.3.1) The above required centrifugal charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, that the pump develops a differential pressure of greater than or equal to 2400 psid when tested pursuant to Technical Specification 4.0.5.

BASES

16.1.2.3.2 See Section 16.1.2.1.2

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

16.1.2.4 (3.1.2.4) At least two centrifugal charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

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

SURVEILLANCE REQUIREMENTS

16.1.2.4.1 (4.1.2.4) At least two centrifugal charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, that the pump develops a differential pressure of greater than or equal to 2400 psid when tested pursuant to Technical Specification 4.0.5.

BASES

16.1.`.4.2 See Section 16.1.2.1.2

*The provisions of Technical Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Technical Specification 4.5.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

16.1.2.5 (3.1.2.5) As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 2968 gallons,

2) Between 7000 and 7700 ppm of boron, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 55,416 gallons,

2) A minimum boron concentration of 2400 ppm, and

3) A minimum solution temperature of 37°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

16.1.2.5.1 (4.1.2.5) The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 37°F.

BASES

16.1.2.5.2 See Section 16.1.2.1.2

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

16.1.2.6 (3.1.2.6) As a minimum, the following borated water sources shall be OPERABLE as required by USAR Section 16.1.2.2 for MODES 1, 2, and 3 and one of the following borated water sources shall be OPERABLE as required by USAR Section 16.1.2.1 for MODE 4:

a. A Boric Acid Storage System with:

1) A minimum contained borated water volume of 17,658 gallons,

2) Between 7000 and 7700 ppm of boron, and

3) A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of 394,000 gallons

2) Between 2400 and 2500 ppm of boron,

3) A minimum solution temperature of 37°F, and

4) A maximum solution temperature of 100°F.

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

ACTION:

a. With the Boric Acid Storage System inoperable and being used as one of the above required borated water sources in MODE 1, 2 or 3, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1.3% $\Delta k/k$ at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable in MODE 1, 2, or 3, 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.

c. With no borated water source OPERABLE in MODE 4, restore one borated water source to OPERABLE status within 6 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

16.1.2.6.1 (4.1.2.6) Each required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) Verifying the Boric Acid Storage System solution temperature when it is the source of borated 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.

BASES

16.1.2.6.2 See Section 16.1.2.1.2

16.1.3 (3/4.1.3) MOVABLE CONTROL ASSEMBLIES

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

16.1.3.1 (3.1.3.3) One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within + 12 steps for each shutdown or control rod not fully inserted.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

16.1.3.1.1 (4.1.3.3) Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicator agrees with the demand position indicator within 12 steps when exercised over the full range of rod travel at least once per 18 months.

BASES

16.1.3.1.2 See Technical Specification Bases 3/4 1.3.

*With the Reactor Trip System breakers in the closed position.
#See Special Test Exception in Section 16.10.2.

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

16.1.3.2 (3.1.3.4) The individual full-length shutdown and control rod drop time from the physical fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the rod drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

16.1.3.2.1 (4.1.3.4) See Technical Specification 4.1.3.1.3

BASES

16.1.3.2.2 See Technical Specification Bases 3/4.1.3

16.2 INTENTIONALLY BLANK

16.3 (3/4.3) INSTRUMENTATION

16.3.1 (3/4.3.3) MONITORING INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

16.3.1.1 (3.3.3.2) The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System,

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$

ACTION:

a. With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions.

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

SURVEILLANCE REQUIREMENTS

16.3.1.1.1 (4.3.3.2) The Movable Incore Detection System shall be demonstrated OPERABLE at least once per 24 hours by normalizing each detector output when required for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$

BASES

16.3.1.1.2 The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q^M(X,Y,Z)$ or $F_{\Delta H}^M(X,Y)$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Neutron Flux channel is inoperable.

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

16.3.1.2 (3.3.3.3) The seismic monitoring instrumentation shown in Table 16.3-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

16.3.1.2.1.a (4.3.3.3.1) Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 16.3-2.

16.3.1.2.1.b (4.3.3.3.2) Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Technical Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

BASES

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

TABLE 16.3-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Peak Recording Accelerographs		
a. Radwaste Base Slab	+ 1.0g	1
b. Control Room	+ 1.0g	1
c. ESW Pump Facility	+ 1.0g	1
d. Ctmt Structure	+ 2.0g	1
e. Auxiliary Bldg. SI Pump Suctions	+ 1.0g	1
f. SGB Piping	+ 5.0g	1
g. SGC Support	+ 1.0g	1
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)		
a. Ctmt. Base Slab	+ 1.0g	1
b. Ctmt. Oper. Floor	+ 1.0g	1
c. Reactor Support	+ 1.0g	1
d. Aux. Bldg. Base Slab	+ 1.0g	1
e. Aux. Bldg. Control Room Air Filter	+ 1.0g	1
f. Free Field	+ 0.5g	1
3. Triaxial Response-Spectrum Recorder (Passive)		
Ctmt. Base Slab	+ 1.0g	1

4. Triaxial Seismic Switches	ACCELERATION LEVEL			
	North	East	Vertical	
a. OBE Ctmt. Base Slab	0.06g	0.06g	0.06g	1
b. SSE Ctmt. Base Slab	0.15g	0.15g	0.16g	1
c. OBE Ctmt. Oper. Fl.	0.07g	0.07g	0.07g	1
d. SSE Ctmt. Oper. Fl.	0.16g	0.17g	0.16g	1
e. System Trigger	0.01g	0.01g	0.01g	1

TABLE 16.3-2

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Peak Recording Accelerographs			
a. Radwaste Base Slab	N.A.	R	N.A.
b. Control Room	N.A.	R	N.A.
c. ESW Pump Facility	N.A.	R	N.A.
d. Ctmt Structure	N.A.	R	N.A.
e. Auxiliary Bldg. SI Pump Suction	N.A.	R	N.A.
f. SGB Piping	N.A.	R	N.A.
g. SGC Support	N.A.	R	N.A.
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)			
a. Ctme. Base Slab	M	R	SA
b. Ctme. Oper. Floor	M	R	SA
c. Reactor Support	M	R	SA**
d. Aux. Bldg. Base Slab	M	R	SA**
e. Aux. Bldg. Control Room Filters	M	R	SA**
f. Free Field	M	R	SA**
3. Triaxial Response-Spectrum Recorder (Passive)			
Ctmt. Base Slab	N.A.	R	N.A.*
4. Triaxial Seismic Switches			
a. OBE Ctmt. Base Slab	M	R	SA
b. SSE Ctmt. Base Slab	M	R	SA
c. OBE Ctmt. Oper. Fl.	M	R	SA
d. SSE Ctmt. Oper. Fl.	M	R	SA
e. System Trigger	M	R	SA

*Checking at the Main Control Board Annunciation for contact closure output in the Control Room shall be performed at least once per 184 days.

**The Bi-stable Trip Setpoint need not be determined during the performance of an ANALOG CHANNEL OPERATIONAL TEST.

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

16.3.1.3 (3.3.3.4) The meteorological monitoring instrumentation channels in Table 16.3-3 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.9.[illegible] within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

16.3.1.3.1 (4.3.3.4) Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 16.3-4.

BASES

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

TABLE 16.3-3

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. Wind Speed	Nominal Elev. 10m	1
	Nominal Elev. 60m	1
2. Wind Direction	Nominal Elev. 10m	1
	Nominal Elev. 60m	1
3. Air Temperature - ΔT	Nominal Elev. 10m-60m	1

TABLE 16.3-4

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Nominal Elev. 10m	D	SA
b. Nominal Elev. 60m	D	SA
2. Wind Direction		
a. Nominal Elev. 10m	D	SA
b. Nominal Elev. 60m	D	SA
3. Air Temperature - ΔT		
a. Nominal Elev. 10-60m	D	SA

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

16.3.1.4 (3.3.3.6) The accident monitoring instrumentation channels shown in Table 16.3-5 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 16.3-5, 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 the unit vent-high range noble gas monitor, less than the Minimum Channels OPERABLE requirements of Table 16.3-5, 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 the unit vent-high range noble gas monitor less than the Minimum Channels OPERABLE requirements of Table 16.3-5, initiate an alternate method of monitoring the appropriate parameter(s) within 72 hours and either restore the 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. The provisions of Technical Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

16.3.1.4.1 (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 16.3-6.

BASES

16.3.1.4.2 See Technical Specification Bases 3/4.3.3.

TABLE 16.3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure - Extended Range	2	1
2. Safety Valve Position Indicator	1/Valve	1/Valve
3. Unit Vent - High Range Noble Gas Monitor	N.A.	1

TABLE 16.3-6

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Extended Range	M	R
2. Safety Valve Position Indicator	M	N.A
3. Unit Vent - High Range Noble Gas Monitor	M	R

INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

16.3.1.5 (3.3.3.9) The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more Loose-Part Detection System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Technical Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

16.3.1.5.1 (4.3.3.9) Each channel of the Loose-Part Detection System shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST except for verification of Setpoint at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

BASES

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

INSTRUMENTATION

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

16.3.1.6 (3.3.3.11) The explosive gas monitoring instrumentation channels shown in Table 16.3-7 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Section 16.11.2 are not exceeded.

APPLICABILITY: As shown in Table 16.3-7.

ACTION:

a. With an explosive gas monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, declare the channel inoperable and take the ACTION shown in Table 16.3-7.

b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 16.3-7. Restore the inoperable instrumentation to OPERABLE status within 30 days, and, if unsuccessful, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 to explain why this inoperability was not corrected in a timely manner.

c. The provisions of Technical Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

16.3.1.6.1 (4.3.3.11) Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 16.3-8.

BASES

16.3.1.6.2 Intentionally Blank

TABLE 16.3-7

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System			
a. Hydrogen Monitor	1/Recombiner	**	1
b. Oxygen Monitor	2/Recombiner	**	2

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.

ACTION 2 - With the Outlet Oxygen Monitor channel inoperable, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both oxygen channels or both the inlet oxygen and inlet hydrogen channels inoperable, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

TABLE 16.3-8

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System				
a. Inlet Hydrogen Monitor	D	Q(1)	M	**
b. Outlet Hydrogen Monitor	D	Q(1)	M	**
c. Inlet Oxygen Monitor	D	Q(2)	M	**
d. Outlet Oxygen Monitor	D	Q(3)	M	**

TABLE NOTATIONS

** During WASTE GAS HOLDUP SYSTEM operation.

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

a. One volume percent hydrogen, balance nitrogen and

b. Four volume percent hydrogen, balance nitrogen.

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

a. One volume per ent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

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

a. 10 ppm by volume oxygen, balance nitrogen, and

b. 80 ppm by volume oxygen, balance nitrogen.

INSTRUMENTATION

16.3.2 (3/4.3.4) TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

16.3.2.1 (3.3.4) At least one Turbine Overspeed Protection System shall be OPERABLE.

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

ACTION:

a. With one stop valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lines or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

16.3.2.1.1a (4.3.4.1) The provisions of Technical Specification 4.0.4 are not applicable.

16.3.2.1.1b (4.3.4.2) The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position:
   1) Four high pressure turbine stop valves,
   2) Six low pressure turbine reheat stop valves, and
   3) Six low pressure turbine reheat intercept valves.

b. At least once per 31 days by cycling each of the four high pressure main turbine governor valves through at least one complete cycle from the running position;

c. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position;

d. At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems; and

e. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

*Not applicable in MODE 2 or 3 with all main steam line isolation valves and associated bypass valves in the closed position and all other steam flow paths to the turbine isolated.

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

16.4 (3/4.4) REACTOR COOLANT SYSTEM

16.4.1 (3/4.4.2) SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

16.4.1.1 (3.4.2.1) A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig $\pm$ 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

16.4.1.1.1(4.4.2.1) No additional requirements other than those required by Technical Specification 4.0.5.

BASES

16.4.1.1.2 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. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

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

16.4.2 (3/4.4.5) STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

16.4.2.1 (3.4.5) Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With one or more steam generators inoperable, restore the inoperable steam generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

16.4.2.1.1.a (4.1.5.0) Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Technical Specification 4.0.5.

16.4.2.1.1.b (4.4.5.1) Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 16.4-1.

16.4.2.1.1.c (4.4.5.2) Steam Generator Tube Sample Selection Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 16.4-2. The inservice inspection of steam generator tubes shall be performed at the Frequencies specified in Section 16.4.2.1.1.d (4.4.5.3) and the inspected tubes shall be verified acceptable per the acceptance criteria of Section 16.4.2.1.1.e (4.4.5.4). The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

(a) Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

(b) The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

SURVEILLANCE REQUIREMENTS (Continued)

1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Section 16.4.2.1.5a.8 (4.4.5.4a.8)) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

(c) The tubes selected as the second and third samples (if required by Table 16.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

(b) If the results of the inservice inspection of a steam generator conducted in accordance with Table 16.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Section 16.4.2.1.1.d (a) (4.4.5.3a).; the interval may then be extended to a maximum of once per 40 months; and

(c) Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 16.4-2 during the shutdown subsequent to any of the following conditions:

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

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

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

4) A main steam line or feedwater line break.

16.4.2.1.1.e (4.4.5.4) Acceptance Criteria

(a) As used in this specification:

1) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Section 16.4.2.1.1.d (c) (4.4.5.3c), above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

SURVEILLANCE REQUIREMENTS (Continued)

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 16.4-2.

16.4.2.1.f (4.4.5.5) Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Technical Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Technical Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission pursuant to Technical Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

BASES

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

Unscheduled inservice inspections are performed on each steam generator following; 1) reactor to secondary tube leaks; 2) seismic occurrence greater than the Operating Basis Earthquake. 3) a loss-of-coolant accident requiring actuation of the Engineered Safety Features, which for this specification is defined to be a break greater than that equivalent to the severance of a 1" inside diameter pipe, or, for a main steamline or feedline, a break greater than that equivalent to a steam generator safety valve failing open: to ensure that steam generator tubes retain sufficient integrity for continued operation. Transients less severe than these do not require inspections because the resulting stresses are well within the stress criteria established by Regulatory Guide 1.121 which unplugged steam generator tubes must be capable of withstanding.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

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

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.9.2 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations. tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

TABLE 4 4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One1			One[1]	One[2]	One[3]

TABLE NOTATIONS

1. The Inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent actions should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per Steam Generator	C-1	None	N. A.	N. A.	N. A.	N. A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N. A.	N. A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform Action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N. A.	N. A.
	C-3	Inspect all tubes in this S G plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to 50.72(b)(2) of 10 CFR Part 50	All other S. G.s are C-1	None	N. A.	N. A.
			Some S. G.s C-2 but no additional S. G. is C-3	Perform action for C-2 result of second sample	N. A.	N. A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Notification to NRC pursuant to 50.72(b)(2) of 10 CFR Part 50	N. A.	N. A.

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during inspection

REACTOR COOLANT SYSTEM

16.4.3 (3/4.4.7) CHEMISTRY

LIMITING CONDITION FOR OPERATION

16.4.3.1 (3.4.7) The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 16.4-3.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

16.4.3.1.1 (4.4.7) The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 16.4-4.

BASES

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

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

TABLE 16.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

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

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

TABLE 16.4-4

REACTOR COOLANT SYSTEM

CHEMISTRY SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least once per 72 hours
Chloride	At least once per 72 hours
Fluoride	At least once per 72 hours

*Not required with T_{avg} less than or equal to 250°F

16.4.4 (3/4.4.9) PRESSURE/TEMPERATURE LIMITS

PRESSURIZER

LIMITING CONDITION FOR OPERATION

16.4.4.1 (3.4.9.2) The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

e. A maximum spray water temperature differential of 583°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

16.4.4.1.1 (4.4.9.2) The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

BASES

16.4.4.1.2 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.

The pressurizer heatup and cool rates shall not exceed 100°F/h and 200°/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 583°F.

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

Also see Technical Specification Bases 3/4.4.9.

REACTOR COOLANT SYSTEM

16.4.5 (3/4.4.10) STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

16.4.5.1 (3.4.10) The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Section 16.4.5.1.1 (4.4.10).

APPLICABILITY: All MODES.

ACTION:

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

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

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

d. The provisions of Technical Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

16.4.5.1.1 (4.4.10) In addition to the requirements of Technical Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. (See Technical Specification 6.8.5)

BASES

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

Components of the Reactor Coolant System were designed to provide access to permit inservice inspection sin accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

16.4.6 (3/4.4.11) REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

16.4.6 (3.4.11) At least one reactor vessel head vent path consisting of at least two valves in series powered from emergency busses shall be OPERABLE and closed.

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

ACTION:

With the above reactor vessel head vent path inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

16.4.6.1.1 (4.4.11) Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING, and

c. Verifying flow through the reactor vessel head vent paths during venting during COLD SHUTDOWN or REFUELING.

BASES

16.4.6.1.2 Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of a reactor vessel head vent path ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

16.5 INTENTIONALLY BLANK

CONTAINMENT SYSTEMS

16.6 (3/4.6) PRIMARY CONTAINMENT

16.6.1 (3/4.6.1) CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

16.6.1.1 (3.6.1.2) Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:
   1) Less than or equal to L_a, 0.20% by weight of the containment air per 24 hours at P_a, 48 psig, or
   2) Less than or equal to L_t, 0.020% by weight of the containment air per 24 hours at P_t, 24 psig.

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

APPLICABILITY: MODES 1, 2, 3, and 40

ACTION:

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

b. If the Reactor Coolant System temperature is above 200 degrees F, with the measured combined leakage rate for all penetrations and valves subject to Types B and C test exceeding 0.60 L_a,

   1) Restore the combined leakage rate to less than 0.60 L_a within 4 hours by one of the following methods:

      a) Repairing the failed containment isolation component, or
      b) Isolating the penetration containing the failed component by closing and the deactivating one automatic valve, or
      c) Isolating the penetration containing the failed component by closing one manual valve, or
      d) Isolating the penetration containing the failed component by using a blind flange.

   2) If the combined leakage rate is not restored to less than 0.60 L_a within 4 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

SURVEILLANCE REQUIREMENTS

16.6.1.1.1 (4.6.1.2) The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

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

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t at which time the above test schedule may be resumed:

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

1) Confirms the accuracy of the test by verifying that the supplemental test result, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a or 0.25 L_t,

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

3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a or 0.75 L_t and 1.25 L_t.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, 48 psig, at intervals no greater than 24 months except for tests involving:

1) Air locks,

2) Purge supply and exhaust isolation valves with resilient material seals, and

3) Valves pressurized with fluid from a seal system.

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Technical Specification 4.6.1.7.2 and 4.6.1.7.4, as applicable;

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (53 psig), and the seal system capacity is adequate to maintain system pressure for at least 30 days; and

h. The provisions of Technical Specification 4.0.2 are not applicable.

BASES

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

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

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

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

16.6.1.2 (3.6.1.6) The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Section 16.6.1.2.1 (4.6.1.6).

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

ACTION:

a. With the abnormal degradation indicated by the conditions in Section 16.6.1.2.1a.4 (4.6.1.6.1a.4), restore the tendons to the required level of integrity or verify that containment integrity is maintained within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Technical Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the indicated abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Section 16.6.1.2.1 (4.6.1.6), restore the containment vessel to the required level of integrity or verify that containment integrity is maintained within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Technical Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Technical Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

16.6.1.2.1.a (4.6.1.6.1) Containment Vessel Tendons. The structural integrity of the prestressing tendons of the containment vessel shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The structural integrity of the tendons shall be demonstrated by:

(a) Determining that a random but representative sample of at least 11 tendons (4 inverted U and 7 hoop) each have an observed lift-off force within the predicted limits established for each tendon. For each subsequent inspection one tendon from each group (1 inverted U and 1 hoop) shall be kept unchanged to develop a history and to correlate the observed data. The procedure of inspection and the tendon acceptance criteria shall be as follows:

SURVEILLANCE REQUIREMENTS (Continued)

1. If the measured prestressing force of the selected tendon in a group lies above the prescribed lower limit, the lift-off test is considered to be a positive indication of the sample tendon's acceptability,

2. If the measured prestressing force of the selected tendon in a group lies between the prescribed lower limit and 90% of the prescribed lower limit, two adjacent (accessible) tendons, one on each side of this tendon shall be checked for their prestressing forces. If the prestressing forces of these two tendons are above 95% of the prescribed lower limits for the tendons, all three tendons shall be restored to the required level of integrity, and the tendon group shall be considered as acceptable. If the measured prestressing force of any two tendons falls below 95% of the prescribed lower limits of the tendons, additional lift-off testing shall be done to detect the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the containment structure,

3. If the measured prestressing force of any tendon lies below 90% of the prescribed lower limit, the defective tendon shall be completely detensioned and additional lift-off testing shall be done so as to determine the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the containment structure,

4. If the average of all measured prestressing forces for each group (corrected for average condition) is found to be less than the minimum required prestress level at the anchorage locations for that group, the condition shall be considered as abnormal degradation of the containment structure,

5. If from consecutive surveillances the measured prestressing forces for the same tendon or tendons in a group indicate a trend of prestress loss larger than expected and the resulting prestressing forces will be less than the minimum required for the group before the next scheduled surveillance, additional lift-off testing shall be done so as to determine the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the containment structure, and

6. Unless there is abnormal degradation of the containment vessel during the first three inspections, the sample population for subsequent inspections shall include at least 6 tendons (3 hoop, 3 inverted U).

SURVEILLANCE REQUIREMENTS (Continued)

(b) Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group. A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire sample (which shall include the broken wire if so identified) that:

1. The tendon wires are free of corrosion, cracks, and damage, and

2. A minimum tensile strength of 240 ksi (guaranteed ultimate strength of the tendon material) exists for at least three wire samples (one from each end and one at mid-length) cut from each removed wire.

Failure to meet the requirements of Section 16.6.1.2.1.a (b) (4.6.1.6.1.b) shall be considered as an indication of abnormal degradation of the containment structure.

(c) Performing tendon retensioning of those tendons detensioned for inspection to at least the force level recorded prior to detensioning or the predicted value, whichever is greater, with the tolerance within minus zero to plus 6%, but not to exceed 70% of the guaranteed ultimate tensile strength of the tendons. During retensioning of these tendons the changes in load and elongation shall be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 10% from that recorded during the installation, an investigation shall be made to ensure that the difference is not related to wire failures or slip of wires in anchorages. This condition shall be considered as an indication of abnormal degradation of the containment structure.

(d) Verifying the OPERABILITY of the sheathing filler grease by assuring:

1. There are no changes in the presence or physical appearance of the sheathing filler-grease including the presence of free water,

2. Amount of grease replaced does not exceed 5% of the net duct volume, when injected at $\pm$ 10% of the specified installation pressure,

3. Minimum grease coverage exists for the different parts of the anchorage system,

4. During general visual examination of the containment external surface, that grease leakage that could affect containment integrity is not present, and

SURVEILLANCE REQUIREMENTS (Continued)

5. The chemical properties of the filler material are within the tolerance limits specified as follows:

Water Content	0 - 10% by dry weight
Chlorides	0 - 10 ppm
Nitrates	0 - 10 ppm
Sulfides	0 - 10 ppm
Reserved Alkalinity	>0

Failure to meet the requirements of Section 16.6.1.2.1.a (d) (4.6.1.6.1.d) shall be considered as an indication of abnormal degradation of the containment structure.

16.6.1.2.1.b (4.6.1.6.2) End Anchorages and Adjacent Concrete Surfaces. As an assurance of the structural integrity of the containment vessel, tendon anchorage assembly hardware (such as bearing plates, stressing washers, wedges, and buttonheads) of all tendons selected for inspection shall be visually examined. Tendon anchorages selected for inspection shall be visually examined to the extent practical without dismantling the load bearing components of the anchorages. Bottom grease caps of all vertical tendons shall be visually inspected to detect grease leakage or grease cap deformations. The surrounding concrete shall also be checked visually for indication of any abnormal condition. The frequency of this surveillance shall be in accordance with Section 16.6.1.2.1 (4.6.1.6.1). Significant grease leakage, grease cap deformation or abnormal concrete condition shall be considered as an indication of abnormal degradation of the containment structure.

16.6.1.2.1.c (4.6.1.6.3) Containment Vessel Surfaces. The exterior surface of the containment shall be visually examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 sq. ft. or more, other surface deterioration or disintegration, or grease leakage, each of which shall be considered as evidence of abnormal degradation of structural integrity of the containment. This inspection shall be performed prior to the Type A containment leakage rate test.

(See Technical Specification 6.8.5)

BASES

16.6.1.2.2 This limitation ensures that the structural integrity of the containment will be maintained in accordance with safety analysis requirements for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 50.4 psig in the event of a steam line break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

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

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

16.7 (3/4.7) PLANT SYSTEMS

16.7.1 (3/4.7.2) STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

16.7.1.1 (3.7.2) The temperatures of both the reactor and secondary coolants in the steam generator shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above section not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 200 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

16.7.1.1.1 (4.7.2) The pressure in each side of the steam generator shall be determined to be less than 200 psig at least once per hour when the temperature of either the reactor or secondary coolant is less than 70°F.

BASES

16.7.1.1.2 The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70 °F and 200 psia are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

PLANT SYSTEMS

16.7.2 (3/4.7.8) SNUBBERS

LIMITING CONDITION FOR OPERATION

16.7.2.1 (3.7.8) All snubbers shall be OPERABLE. The only snubbers excluded from the requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Section 16.7.2.1.1g (4.7.8g) on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

16.7.2.1.1 (4.7.8) Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Technical Specification 4.0.5.

a. Inspection Types

As used in this specification type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 16.7-1. The visual inspection interval for each type snubber shall be determined based upon the criteria provided in Table 16.7-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Operating License Amendment 44.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, and (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; or (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Section 16.7.2.1.1f (4.7.8f). All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to determine system operability with an unacceptable snubber. If operability cannot be justified, the system shall be declared inoperable and the ACTION requirements shall be met.

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

SURVEILLANCE REQUIREMENTS (Continued)

e. Functional Tests
During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers of each type shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Section 16.7.2.1.1f (4.7.8f), an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested, or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 16.7-1 (4.7-1). "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Section 16.7.2.1.1f (4.7.8f). The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on the Figure 16.7-1 (4.7-1). If at any time the point plotted falls in the "Reject" region, all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region or all the snubbers of that type have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2 where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

e. Functional Tests (Continued)

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed rate, or release rate where required, is present in both tension and compression, within the specified range; and

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Service Life Monitoring Program

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

SURVEILLANCE REQUIREMENTS (Continued)

g. Service Life Monitoring Program (Continued)

for the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Section 16.7.2.1.1e (4.7.8e). for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent servic^ and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Technical Specification 6.10.2.

(See Technical Specification 6.8.5)

16.7.2.1.2 All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip, and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer. Snubbers may also be classified and grouped by inaccessible or accessible for visual inspection purposes. Therefore, each snubber type may be grouped for inspection in accordance with accessibility.

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

The visual inspection frequency is based upon maintaining a constant level of snubber protection during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection of each type. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing. Since the visual inspections are augmented by functional testing program, the visual inspection need not be a hands on inspection, but shall require visual scrutiny sufficient to assure that fasteners or mountings for connecting the snubbers to supports or foundations shall have no visible bolts, pins or fasteners missing, or other visible signs of physical damage such as cracking or loosening.

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

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

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

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

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

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

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

TABLE 16.7-1

SNUBBER VISUAL INSPECTION INTERVAL

NUMBER OF UNACCEPTABLE SNUBBERS

Population per Category (Notes 1 and 2)	Column A Extend Interval (Extend 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber category shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, categories must be determined and documented before any inspection and that determination shall be the basis upon which to determine the next inspection interval for that category.

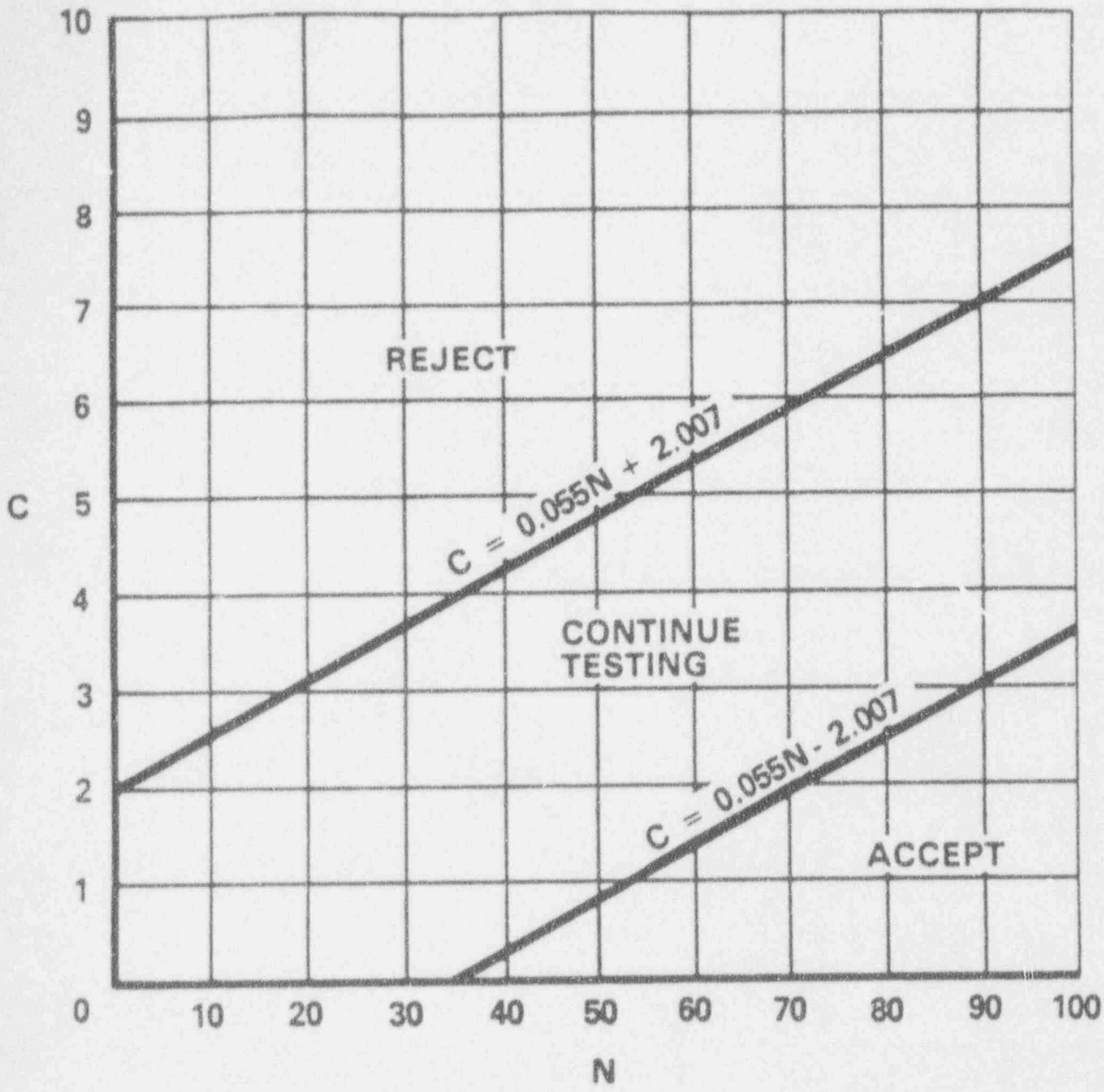
Note 2: Interpolation between population per category and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, and C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column 8 but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Column B and C.

Note 6: The provisions of Technical Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.



16.7-1

FIGURE ~~4.7-1~~

SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

16.7.3 (3/4.7.9) SEALED SOURCE CON[illegible]INATION

LIMITING CONDITION FOR OPERATION

16.7.3.1 (3.7.9) Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma-emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

16.7.3.1.1.a (4.7.9.1) Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

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

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

16.7.3.1.1.b (4.7.9.2) Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

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

   2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use, and

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

16.7.3.1.1.c (4.7.9.3) Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.

BASES

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

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

16.7.4 (3/4.7.12) AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

16.7.4.1 (3.7.12) The temperature limit of each area given in Table 16.7-2 shall not be exceeded for more than 8 hours or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION

a. With one or more areas exceeding the temperature limit(s) shown in Table 16.7-2 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Technical Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Technical Specifications 3.0.3 and 3.0.4 are not applicable.

b. With one or more areas exceeding the temperature limit(s) shown in Table 16.7-2 by more than 30°F, prepare and submit a Special Report as required by ACTION a. above, and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

16.7.4.1.1 (4.7.12) The temperature in each of the areas shown in Table 16.7-2 shall be determined to be within its limit at least once per 12 hours.

(See Technical Specification 6.8.5)

BASES

16.7.4.1.2 The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ±3°F.

TABLE 16.7-2

AREA TEMPERATURE MONITORING

AREA	MAXIMUM TEMPERATURE LIMIT (°F)
1. ESW Pump Room A	119
2. ESW Pump Room B	119
3. Auxiliary Feedwater Pump Room A	119
4. Auxiliary Feedwater Pump Room B	119
5. Turbine Driven Auxiliary Feedwater Pump Room	147
6. ESF Switchgear Room I	87
7. ESF Switchgear Room II	87
8. RHR Pump Room A	119
9. RHR Pump Room B	119
10. CTMT Spray Pump Room A	119
11. CTMT Spray Pump Room B	119
12. Safety Injection Pump Room A	119
13. Safety Injection Pump Room B	119
14. Centrifugal Charging Pump Room A	119
15. Centrifugal Charging Pump Room B	119
16. Electrical Penetration Room A	101
17. Electrical Penetration Room B	101
18. Component Cooling Water Room A	119
19. Component Cooling Water Room B	119
20. Diesel Generator Room A	119
21. Diesel Generator Room B	119
22. Control Room	84

16.8 (3/4.8) ELECTRICAL POWER SYSTEMS

16.8.1 (3/4.8.4) ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

16.8.1.1 (3.8.4.1) For each containment penetration provided with a penetration conductor overcurrent protective device(s), each device shall be OPERABLE.

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

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Technical Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

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

SURVEILLANCE REQUIREMENTS

16.8.1.1.1 (4.8.4.1) Protective devices required to be OPERABLE as containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE.

a. At least once per 18 months:

1) By verifying that the 13.8 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers, and performing the following:

a) A CHANNEL CALIBRATION of the associated protective relays,

b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed, and

SURVEILLANCE REQUIREMENTS (Continued)

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal Setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

BASES

16.8.1.1.2 Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

A list of containment penetration conductor overcurrent protective devices whose circuit limiting fault current exceeds the penetration rating, with information of location and size and equipment powered by the protected circuit, is available at the plant site in accordance with Section 50.71(c) of 10 CFR Part 50. The addition or deletion of any containment penetration conductor overcurrent protective device would be made in accordance with Section 50.59 of 10 CFR Part 50.

16.9 (3/4.9) REFUELING OPERATIONS

16.9.1 (3/4.9.5) COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

16.9.1.1 (3.9.5) Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

16.9.1.1.1 (4.9.5) Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

BASES

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

16.9.2 (3/4.9.6) REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

16.9.2.1 (3.9.6) The refueling machine shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The refueling machine used for movement of fuel assemblies having:

1) A minimum capacity of 4800 pounds,

2) Automatic overload cutoffs with the following Setpoints:

a) Primary - less than or equal to 250 pounds above the indicated suspended weight for wet conditions and less than or equal to 350 pounds above the indicated suspended weight for dry conditions, and

b) Secondary - less than or equal to 150 pounds above the primary overload cutoff.

3) An automatic load reduction trip with a Setpoint of less than or equal to 250 pounds below the suspended weight for wet conditions or dry conditions.

b. The auxiliary hoist used for latching and unlatching drive rods and thimble plug handling operations having:

1) A minimum capacity of 3000 pounds, and

2) A 1000-pound load indicator which shall be used to monitor lifting loads for these operation.

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

ACTION:

With the requirements for refueling machine and/or auxiliary hoist OPERABILITY not satisfied, suspend use of any inoperable refueling machine crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

16.9.2.1.1.a (4.9.6.1) The refueling machine used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior

SURVEILLANCE REQUIREMENTS (Continued)

to the movement of fuel assemblies in the reactor vessel by performing a load test of at least 125% of the secondary automatic overload cutoff and demonstrating an automatic load cutoff when the refueling machine load exceeds the Setpoints of Section 16.9.2.1a.2.) and by demonstrating an automatic load reduction trip when the load reduction exceeds the Setpoint of Section 16.9.2.1a.3.

16.9.2.1.1.b (4.9.6.2) Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the movement of drive rods within the reactor vessel by performing a load test of at least 1250 pounds.

BASES

16.9.2.1.2 The OPERABILITY requirements for the refueling machine and auxiliary hoist ensure that: (1) manipulator cranes will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

REFUELING OPERATIONS

16.9.3 (3/4.9.7) CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

LIMITING CONDITION FOR OPERATION

16.9.3.1 (3.9.7) Loads in excess of 2250 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage facility.

APPLICABILITY: With fuel assemblies in the spent fuel storage facility.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

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

SURVEILLANCE REQUIREMENTS

16.9.3.1.1 (4.9.7) Crane interlocks and physical stops which prevent crane travel with loads in excess of 2250 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

BASES

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

16.9.4 (3/4.9.10)WATER LEVEL - REACTOR VESSEL

CONTROL RODS

LIMITING CONDITION FOR OPERATION

16.9.4.1 (3.9.10.2) At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor pressure vessel.

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

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of control rods within the pressure vessel.

SURVEILLANCE REQUIREMENTS

16.9.4.1.1 (4.9.10.2) The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of control rods within the reactor vessel.

BASES

16.9.4.1.2 See Technical Specification Bases 3/4.9.10

16.10 (3/4.10) SPECIAL TEST EXCEPTIONS

16.10.1 (3/4.10.1) SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

16.10.1.1 (3.10.1) The SHUTDOWN MARGIN requirement of Section 16.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Section 16.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Section 16.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

16.10.1.1.1.a (4.10.1.1) The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

16.10.1.1.1.b (4.10.1.2) Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Section 16.1.1.1.

BASES

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

SPECIAL TEST EXCEPTIONS

16.10.2 (3/4.10.5) POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

16.10.2.1 (3.10.5) The limitations of Section 16.1.3.1 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided only one shutdown or control bank is withdrawn from the fully inserted position at a time.

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements and during surveillance of digital rod position indicators for OPERABILITY.

ACTION:

With the Position Indication System inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

16.10.2.1.1 (4.10.5) The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

BASES

16.10.2.1.2 This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements.

16.11 (3/4.11) RADIOACTIVE EFFLUENTS

16.11.1 LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

16.11.1.1 (3.11.1.4) The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 150 Curies, excluding tritium and dissolved or entrained noble gases.

a. Reactor Makeup Water Storage Tank,
b. Refueling Water Storage Tank,
c. Condensate Storage Tank, and
d. Outside temporary tanks, excluding demineralizer vessels and liners being used to solidify radioactive wastes.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Technical Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

16.11.1.1.1 (4.11.1.4) The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added and within 7 days following any addition of radioactive material to the tank. (See Technical Specification 6.8.5)

BASES

16.11.1.1.2 The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

16.11.2 EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

16.11.2.1 (3.11.2.5) The concentration of oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 3% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 3% by volumo but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the WASTE GAS HOLDUP SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a. above.

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

SURVEILLANCE REQUIREMENTS

16.11.2.1.1 (4.11.2.5) The concentrations of hydrogen and oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 16.3-6 of Section 16.3.1.6.

(See Technical Specification 6.8.5)

BASES

16.11.2.1.2 This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the WASTE GAS HOLDUP SYSTEM is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

RADIOACTIVE EFFLUENTS

16.11.3 GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

16.11.3.1 (3.11.2.6) The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 2.5 x 10^5 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, and within 48 hours, reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Technical Specification 6.9.1.7.

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

SURVEILLANCE REQUIREMENTS

16.11.3.1.1 (4.11.2.6) The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days when radioactive materials are being added and within 7 days following any addition of radioactive material to the tank. (See Technical Specification 6.8.5)

BASES

16.11.3.1.2 The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.